

FINAL

ENVIRONMENTAL IMPACT STATEMENT

**Management of
Commercially Generated
Radioactive Waste**

Volume 1



October 1980

**U.S. Department of Energy
Assistant Secretary for Nuclear Energy
Office of Nuclear Waste Management
Washington, D.C. 20545**

FOREWORD

In his February 12, 1980, message to Congress, the President of the United States announced a comprehensive program for management of radioactive waste. With regard to waste disposal, the President said:

". . . for disposal of high-level radioactive waste, I am adopting an interim planning strategy focused on the use of mined geologic repositories capable of accepting both waste from reprocessing and unprocessed commercial spent fuel. An interim strategy is needed since final decisions on many steps which need to be taken should be preceded by a full environmental review under the National Environmental Policy Act. In its search for suitable sites for high-level waste repositories, the Department of Energy has mounted an expanded and diversified program of geologic investigations that recognizes the importance of the interaction among geologic setting, repository host rock, waste form, and other engineered barriers on a site-specific basis. Immediate attention will focus on research and development and on locating and characterizing a number of potential repository sites in a variety of different geologic environments with diverse rock types. When four to five sites have been evaluated and found potentially suitable, one or more will be selected for development as a licensed, full-scale repository."

In an accompanying Fact Sheet issued by the White House Press Secretary it was noted that the President will reexamine this interim strategy and decide whether any changes need to be made following completion of the necessary environmental reviews as required by the National Environmental Policy Act (NEPA). Issuance of this environmental impact statement (EIS) is intended to serve as a basis for that reexamination.

In keeping with the mandate of NEPA, this EIS analyzes the significant environmental impacts that could occur if various technologies for management and disposal of high-level and transuranic wastes from commercial nuclear power reactors were to be developed and implemented. This EIS will serve as the environmental input for the decision on which technology, or technologies, will be emphasized in further research and development activities in the commercial waste management program.

The action proposed in this EIS is to 1) adopt a national strategy to develop mined geologic repositories for disposal of commercially generated high-level and transuranic radioactive waste (while continuing to examine seabed and very deep hole disposal as potential backup technologies) and 2) conduct an R&D program to develop such facilities and the necessary technology to ensure the safe long-term containment and isolation of these wastes.

The Department has considered in this Statement:

- Development of conventionally mined deep geologic repositories for disposal of spent fuel from nuclear power reactors and/or radioactive fuel reprocessing wastes.^(a)
- Balanced development of several alternative disposal methods.
- No waste disposal action.

Prior to announcing his national waste management program, the President received recommendations on the program from an Interagency Review Group whose report was issued in April 1979. In their report, the Interagency Review Group analyzed a number of possible strategies for the program of high-level waste disposal. These strategies differed with regard to the number of diverse sites that should be examined in a geologic disposal program prior to construction of a facility and in one case discussed the implementation of technologies other than mined geologic repositories.

This EIS has not specifically examined the strategies reviewed by the Interagency Review Group but the essential differences between them are covered in the comparison of the first two program alternatives considered here. These alternatives have been examined for a number of different scenarios of future nuclear power use and for a range of times for operation of facilities, including those considered by the Interagency Review Group.

A draft of this environmental impact statement--"Management of Commercially Generated Radioactive Waste"--was issued for review and comment as DOE/EIS-0046D on April 20, 1979. Copies were sent to Federal agencies with responsibilities associated with radioactive waste disposal, to governors of all states, and to public interest groups known to have an interest in waste management. Comments were received from the following Federal agencies:

Department of Commerce
 Department of Health, Education and Welfare
 Department of the Interior
 Environmental Protection Agency
 Federal Energy Regulatory Commission
 Nuclear Regulatory Commission
 and from agencies or officials from 17 states.

A total of 219 written communications, incorporating about 2000 comments, were received and considered in preparation of this final Statement.

An impartial Hearing Board, composed of specialists in several fields, was appointed to conduct a series of public hearings on the draft Statement. The board members had not been DOE personnel nor previously involved with the DOE waste management program and were employed specifically to conduct the hearings and evaluate the public concerns. Hearings were held in Washington, D.C.; Chicago, Illinois; Atlanta, Georgia; Dallas, Texas; and San

(a) The Statement does not formally consider radioactive wastes related to defense programs; however, in a generic sense, systems that can safely dispose of commercial radioactive wastes are expected to safely dispose of defense wastes.

Francisco, California. Transcripts of these hearings have been made available in DOE reading rooms.^(a) The Hearing Board issued their report in February 1980 recommending revisions to the draft Statement based upon comments made by members of the public at the hearings and upon evaluations of their own observations.

Summaries of issues raised in written comments, responses to them, and the report of the Hearing Board are included in Volume 3 of this Statement. Changes in the text as a result of the comment process, including hearing testimony, appear throughout the Statement as indexed in Volume 3. The final Statement has been reorganized extensively for improved readability.

Dr. Colin A. Heath, Director, Office of Waste Isolation, Mail Stop B-107, Washington, D.C. 20545, is the responsible Department of Energy manager for this Statement. The Pacific Northwest Laboratory, operated by Battelle Memorial Institute for the Department of Energy, was assigned prime responsibility for preparing the draft and final Statement.

Single copies of this Statement may be obtained by writing:

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(a) The locations of the DOE regional offices, which contain the DOE reading rooms, are provided at the end of this Foreword.

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VOLUME 1CONTENTS

FOREWORD	iii
CHAPTER 1 - SUMMARY	1.1
1.1 THE NEED FOR WASTE MANAGEMENT AND DISPOSAL	1.3
1.2 THE PROGRAMMATIC ALTERNATIVES	1.6
1.3 THE PROPOSED ACTION	1.7
1.3.1 Mined Geologic Disposal of Radioactive Wastes	1.8
1.3.2 An Example Geologic Repository	1.8
1.3.3 Environmental Impacts Associated with Construction and Operation of Example Geologic Repositories	1.10
1.3.3.1. Radiological Impacts	1.10
1.3.3.2 Resource Commitments	1.11
1.3.3.3 Socioeconomic Impacts	1.11
1.3.3.4 Land Use, Ecological Impacts and Other Impacts	1.12
1.3.4 Environmental Impacts in the Long Term	1.13
1.4 ALTERNATIVE ACTION--BALANCED DEVELOPMENT OF ALTERNATIVE DISPOSAL METHODS	1.16
1.4.1 Very Deep Hole Waste Disposal Concept	1.16
1.4.2 Rock Melt Waste Disposal Concept	1.16
1.4.3 Island-based Geologic Disposal Concept	1.17
1.4.4 Subseabed Disposal Concept	1.17
1.4.5 Ice Sheet Disposal Concept	1.18
1.4.6 Well Injection Disposal Concepts	1.18
1.4.7 Transmutation Concept	1.19
1.4.8 Space Disposal Concept	1.19
1.5 NO-ACTION ALTERNATIVE	1.21
1.6 PREDISPOSAL SYSTEMS	1.22
1.6.1 Predisposal System for the Once-Through Cycle	1.22
1.6.2 Predisposal System for the Reprocessing Cycle	1.22

CONTENTS (contd)

1.6.3	Accident Impact Summary for Predisposal Operations	1.22
1.7	ENVIRONMENTAL IMPACTS OF PROGRAMMATIC ALTERNATIVES FOR THE ONCE-THROUGH AND THE REPROCESSING FUEL CYCLE OPTIONS AND VARIOUS NUCLEAR POWER GROWTH ASSUMPTIONS	1.24
1.7.1	System Radiological Impacts	1.26
1.7.2	System Resource Commitments	1.27
1.7.3	Systems Costs	1.29
1.8	CONCLUSIONS	1.31
	REFERENCES FOR CHAPTER 1	1.34
CHAPTER 2	INTRODUCTION	2.1
2.1	RELATIONSHIP TO OTHER WASTE MANAGEMENT DECISIONS	2.1
2.1.1	Mining and Milling	2.2
2.1.2	Uranium Enrichment	2.3
2.1.3	Uranium Fuel Fabrication	2.3
2.1.4	Low-Level Waste	2.3
2.1.5	Spent Fuel Storage	2.3
2.1.6	Transportation	2.4
2.1.7	Alternative Reactor Types	2.4
2.1.8	Wastes From National Defense Activities	2.4
2.1.9	National Plan for Nuclear Waste Management	2.5
2.2	STRUCTURE AND CONTENT OF STATEMENT	2.7
2.3	OTHER DECISIONS CONCERNING DISPOSAL OF COMMERCIAL WASTES	2.10
2.3.1	The DOE's National Environmental Policy Act Implementation	2.12
2.3.1.1	Program Strategy	2.12
2.3.1.2	Site Selection Process	2.12
2.3.1.3	Land Acquisition	2.14
	REFERENCES FOR CHAPTER 2	2.15
CHAPTER 3	DESCRIPTION OF PROGRAM ALTERNATIVES AND BACKGROUND	3.1
3.1	PROPOSED ACTION AND PROGRAM ALTERNATIVES	3.1
3.1.1	Proposed Action	3.4

CONTENTS (contd)

3.1.2	Alternative Action--Parallel Development	3.5
3.1.3	No-Action Alternative	3.6
3.2	BASES FOR THE ANALYSIS	3.7
3.2.1	Nuclear Fuel Cycle Assumptions	3.8
3.2.1.1	Once-Through Fuel Cycle	3.8
3.2.1.2	Reprocessing Fuel Cycle	3.9
3.2.2	Nuclear Power Growth Assumptions	3.12
3.2.3	Resource Commitment Assessment	3.14
3.2.4	Ecological and Atmospheric Impacts	3.15
3.2.5	Radiological Impacts Assessments and Uncertainties	3.15
3.2.6	Socioeconomic Impacts	3.16
3.2.7	Basis for Accident Analysis	3.17
3.2.8	Cost Analysis Bases	3.19
3.2.8.1	Bases for Capital, Operating and Decommissioning Cost Estimates	3.19
3.2.8.2	Bases for Levelized Unit Cost Estimates	3.20
3.2.8.3	Uncertainty Ranges for Cost Calculations	3.21
3.2.8.4	Cost Estimates for Transportation	3.21
3.2.8.5	Research and Development Costs	3.22
3.2.9	Physical Protection Safeguard Requirements Assessment	3.22
	REFERENCES FOR SECTION 3.2	3.25
3.3	NATURALLY OCCURRING RADIATION AND STANDARDS FOR EXPOSURE TO MAN-MADE RADIATION	3.26
3.3.1	Natural Radioactivity and Radiation Dose	3.26
3.3.1.1	Cosmic Radiation	3.27
3.3.1.2	Terrestrial Radioactivity	3.27
3.3.1.3	Summary of Whole-Body Dose	3.29
3.3.2	Applicable Standards for Radiation Exposure Control	3.30
3.3.2.1	Basic Radiation Standards	3.30
3.3.2.2	Other Requirements	3.32

CONTENTS (contd)

REFERENCES FOR SECTION 3.3	3.35
3.4 RISK AND RISK PERSPECTIVES	3.36
3.4.1 Hazard Indices	3.36
3.4.2 Consequence Analysis and Risk Assessment	3.38
3.4.2.1 Disruptive Events	3.39
3.4.2.2 Lithosphere/Atmosphere Transport	3.39
REFERENCES FOR SECTION 3.4	3.42
3.5 NONTECHNICAL ISSUES	3.43
3.5.1 Social Issues	3.43
3.5.2 Institutional Issues	3.47
3.5.2.1 Short-Term Concerns and Institutional Design	3.47
3.5.2.2 Institutions in Long-Term Nuclear Waste Management	3.49
REFERENCES FOR SECTION 3.5	3.51
CHAPTER 4 - PREDISPOSAL SYSTEMS	4.1
4.1 RELATIONSHIP OF PREDISPOSAL OPERATIONS TO DISPOSAL AND PROGRAM ALTERNATIVES	4.1
4.1.1 Predisposal System for the Once-Through Cycle	4.1
4.1.2 Predisposal System for the Reprocessing Cycle	4.4
4.1.3 Predisposal System Relationships to Program Alternatives	4.7
4.2 UNTREATED WASTE CHARACTERIZATION	4.9
4.2.1 Once Through-Cycle Wastes	4.11
4.2.2 Reprocessing Cycle	4.13
4.3 WASTE TREATMENT AND PACKAGING	4.17
4.3.1 Spent Fuel Treatment and Packaging in Once-Through Cycle	4.17
4.3.1.1 Encapsulate Intact Assembly (Example Method)	4.17
4.3.1.2 Chop Fuel Assembly, Voloxidize Fuel, and Encapsulate	4.19
4.3.1.3 Dissolve Fuel and Convert to Glass	4.20
4.3.1.4 Dissolve Fuel for Disposal as a Liquid	4.20
4.3.2 High-Level Liquid Waste Treatment	4.22
4.3.2.1 Chemical Partitioning	4.25

CONTENTS (contd)

4.3.2.2	Glass Waste Forms	4.27
4.3.2.3	Crystalline Waste Forms	4.28
4.3.2.4	Composite Waste Forms	4.30
4.3.2.5	Waste Form Characterization	4.31
4.3.3	TRU Waste Treatment in the Reprocessing Cycle	4.32
4.3.3.1	Fuel Residue Treatment	4.32
4.3.3.2	Failed Equipment and Other Noncombustible Waste Treatment	4.34
4.3.3.3	Combustible and Compactable Waste Treatment	4.36
4.3.3.4	Immobilization of Wet Wastes and Particulate Solids	4.38
4.3.4	Gaseous and Airborne Waste Treatment	4.39
4.3.4.1	Filtration	4.40
4.3.4.2	Gaseous Radionuclide Recovery	4.42
4.3.5	Radionuclide Releases During Waste Treatment and Packaging	4.43
4.3.6	Treated Waste Quantities	4.44
REFERENCES FOR SECTION 4.3	4.47
4.4	WASTE STORAGE	4.51
4.4.1	Spent Fuel Storage	4.51
4.4.1.1	Water Basin Storage of Unpackaged Spent Fuel (Example Method)	4.52
4.4.1.2	Water Basin Storage of Packaged Spent Fuel	4.52
4.4.1.3	Air-Cooled Vault Storage of Packaged Spent Fuel	4.53
4.4.1.4	Dry Well Storage of Packaged Spent Fuel	4.54
4.4.1.5	Surface Cask Storage of Packaged Spent Fuel	4.54
4.4.2	High-Level Waste Storage	4.55
4.4.2.1	Tank Storage of Liquid High-Level Waste	4.55
4.4.2.2	Water Basin Storage of Solidified High-Level Waste (Example Method)	4.57
4.4.2.3	Sealed Cask Storage of Solidified High-Level Waste	4.57
4.4.2.4	Other Solidified High-Level Waste Storage Concepts	4.58
4.4.3	TRU Waste Storage	4.58

CONTENTS (contd)

4.4.3.1	Vault Storage of RH-TRU (Example Method for Drummed RH-TRU)	4.59
4.4.3.2	Dry-Well Storage of RH-TRU (Example Method for Canistered RH-TRU)	4.59
4.4.3.3	Unshielded Indoor Storage of CH-TRU	4.60
4.4.3.4	Outdoor Storage of CH-TRU (Example Method)	4.60
4.4.4	Krypton Storage	4.61
REFERENCES FOR SECTION 4.4		4.62
4.5	WASTE TRANSPORT	4.63
4.5.1	Spent Fuel Transport	4.63
4.5.2	High-Level Waste Transport	4.65
4.5.3	TRU Waste Transport	4.65
4.5.3.1	Fuel Residue Transport	4.66
4.5.3.2	Other TRU Waste Transport	4.66
REFERENCES FOR SECTION 4.5		4.68
4.6	DECOMMISSIONING OF RETIRED FACILITIES	4.69
REFERENCES FOR SECTION 4.6		4.75
4.7	ENVIRONMENTAL IMPACTS OF PREDISPOSAL OPERATIONS	4.76
4.7.1	Environmental Impacts Related to Predisposal Operations for the Once-Through Fuel Cycle	4.76
4.7.1.1	Resource Commitments for Once-Through Fuel Cycle Waste Management	4.76
4.7.1.2	Nonradiological Effluents of Once-Through Fuel Cycle Waste Management	4.78
4.7.1.3	Radiological Effects of Once-Through Fuel Cycle Waste Management	4.78
4.7.1.4	Ecological Effects of Once-Through Fuel Cycle Waste Management	4.79
4.7.1.5	Socioeconomic Impacts of Once-Through Fuel Cycle Waste Management	4.79
4.7.2	Environmental Impacts Related to Predisposal Operations for the Reprocessing Fuel Cycle	4.81
4.7.2.1	Resource Commitments in Reprocessing Fuel Cycle Waste Management	4.81

CONTENTS (contd)

4.7.2.2	Nonradiological Effluents of Reprocessing Fuel Cycle Waste Management	4.83
4.7.2.3	Radiological Effects of Reprocessing Fuel Cycle Waste Management	4.84
4.7.2.4	Ecological Effects of Reprocessing Fuel Cycle Waste Management	4.86
4.7.2.5	Socioeconomic Impacts of Reprocessing Fuel Cycle Waste Management	4.87
REFERENCES FOR SECTION 4.7	4.89
4.8	ACCIDENT IMPACTS FOR PREDISPOSAL OPERATIONS	4.90
4.8.1	Accident Impacts for the Once-Through Cycle	4.90
4.8.1.1	Radiological Impacts from Spent Fuel Transportation Accidents	4.90
4.8.1.2	Radiological Impacts from Unpackaged Spent Fuel Storage Accidents	4.92
4.8.1.3	Radiological Impacts Due to Accidents at a Fuel Packaging Facility	4.92
4.8.1.4	Radiological Impacts from Packaged Spent Fuel Storage Accidents	4.93
4.8.1.5	Non-Radiological Impacts of Accidents in the Once-Through Cycle	4.93
4.8.2	Accident Impacts for the Reprocessing Fuel Cycle	4.94
4.8.2.1	Radiological Impacts from Accidents During the Treatment and Packaging of	4.94
4.8.2.2	Radiological Impacts from Reprocessing Waste Storage Accidents	4.96
4.8.2.3	Radiological Impacts from Reprocessing Waste Transportation Accidents	4.97
4.8.2.4	Non-Radiological Impacts of Accidents in the Reprocessing Cycle	4.98
4.8.3	Radiological Impact Summary for Predisposal Operations Accidents	4.98
REFERENCES FOR SECTION 4.8	4.100
4.9	COST OF PREDISPOSAL OPERATIONS	4.101
4.9.1	Once-Through Fuel Cycle Predisposal Costs	4.101
4.9.2	Reprocessing Fuel Cycle Predisposal Costs	4.103

CONTENTS (contd)

4.9.3	Detailed Predisposal Cost Estimates for Geologic Disposal	4.105
4.9.3.1	Once-Through Fuel Cycle	4.105
4.9.3.2	Reprocessing Fuel Cycle	4.106
4.9.4	Detailed Subsystem Costs for Geologic Disposal	4.108
REFERENCES FOR SECTION 4.9	4.111
4.10	SAFEGUARDS INCLUDING PHYSICAL PROTECTION FOR PREDISPOSAL OPERATIONS	4.112
4.10.1	Safeguards Requirements for the Once-Through Cycle	4.112
4.10.1.1	Spent Fuel Treatment and Packaging Safeguards Requirements	4.112
4.10.1.2	Safeguards Requirements for Spent Fuel Storage	4.112
4.10.1.3	Safeguards Requirements for Transport of Spent Fuel	4.113
4.10.2	Safeguards Requirements for the Reprocessing Fuel Cycle	4.114
4.10.2.1	Safeguards Requirements for the Treatment of Reprocessing Wastes	4.114
4.10.2.2	Safeguards Requirements for Storage of Reprocessing Cycle Wastes	4.115
4.10.2.3	Safeguards Requirements for Transport of Reprocessing Cycle Wastes	4.117
REFERENCES FOR SECTION 4.10	4.118
CHAPTER 5 - GEOLOGIC DISPOSAL		5.1
5.1	DESCRIPTION OF THE GEOLOGIC DISPOSAL CONCEPT	5.1
5.1.1	Factors Relevant to Geologic Disposal	5.2
5.1.1.1	Disposal Media Properties	5.3
5.1.1.2	Generic Basis for Repository Site Selection	5.5
5.1.1.3	Generic Basis for Repository Design	5.6
5.1.2	Engineered Barriers	5.6
5.1.2.1	Engineered Barriers--Waste Package System	5.6
5.1.2.2	Waste Packages Components	5.8
5.1.2.3	Waste Package Development and Assessment	5.10
5.1.2.4	Current Status of Waste Package Development in U.S.	5.11
REFERENCES FOR SECTION 5.1	5.13

CONTENTS (contd)

5.2	STATUS OF TECHNOLOGY AND R&D	5.14
5.2.1	Geologic Site Selection	5.14
5.2.1.1	Methods for Regional Geologic Studies	5.15
5.2.1.2	Methods for Site Analysis	5.15
5.2.2	Waste Package Systems	5.21
5.2.2.1	Waste Form	5.21
5.2.2.2	Materials	5.22
5.2.3	Repository System	5.23
5.2.3.1	Excavation and Underground Development	5.24
5.2.3.2	Thermal Effects	5.24
5.2.3.3	Radiation Effects	5.24
5.2.3.4	Repository Penetration	5.25
5.2.4	Summary	5.25
	REFERENCES FOR SECTION 5.2	5.26
5.3	DESCRIPTION OF THE CONCEPTUAL GEOLOGIC REPOSITORY FACILITIES	5.29
5.3.1	Once-Through Fuel Cycle Repository	5.29
5.3.1.1	Design Bases	5.29
5.3.1.2	Facility Description	5.31
5.3.1.3	Construction	5.34
5.3.1.4	Operations	5.34
5.3.1.5	Retrievability	5.35
5.3.1.6	Decommissioning	5.37
5.3.2	Reprocessing Fuel Cycle Repository	5.37
5.3.2.1	Design Bases	5.37
5.3.2.2	Facility Description	5.39
5.3.2.3	Construction	5.39
5.3.2.4	Operations	5.39
5.3.2.5	Retrievability	5.42
5.3.2.6	Decommissioning	5.42

CONTENTS (contd)

5.3.3	Effect of Waste Age on Repository Capacity	5.42
5.3.4	Regional Repository Concept	5.43
	REFERENCES FOR 5.3	5.45
5.4	ENVIRONMENTAL IMPACTS RELATED TO REPOSITORY CONSTRUCTION AND OPERATION	5.46
5.4.1	Resource Commitments	5.46
5.4.2	Nonradiological Effluents	5.48
5.4.3	Radiological Effects	5.52
5.4.4	Evaluation of Ecological Impacts Related to Repositories	5.53
5.4.4.1	Ecological Effects Related to Repositories in Salt	5.53
5.4.4.2	Ecological Effects for a Repository in Granite	5.54
5.4.4.3	Ecological Effects for a Repository in Shale	5.55
5.4.4.4	Ecological Effects for a Repository in Basalt	5.56
5.4.4.5	Ecological Impacts Related to Repositories for Reprocessing Wastes	5.56
5.4.5	Nonradiological Accidents	5.56
5.4.6	Environmental Effects Related to Repository Operation	5.57
5.4.6.1	Resource Commitments	5.58
5.4.6.2	Nonradiological Effluents	5.58
5.4.6.3	Radiological Releases	5.60
5.4.6.4	Ecological Impacts	5.60
5.4.6.5	Socioeconomic Impacts	5.61
5.4.6.6	Environmental Effects Related to Postulated Radiological Accidents	5.66
5.4.6.7	Radiological Impacts of Operating Accidents on the Work Force	5.69
5.4.6.8	Other Environmental Impacts	5.69
	REFERENCES FOR SECTION 5.4	5.71
5.5	LONG-TERM ENVIRONMENTAL CONSIDERATIONS OF GEOLOGIC DISPOSAL OF RADIOACTIVE WASTES	5.72
5.5.1	Repository Breach by Meteorite	5.74
5.5.2	Breach of Repository by Fault, Fracture, and Flooding	5.80

CONTENTS (contd)

5.5.3	Faulting and Ground-water Intrusion to a Domestic Well	5.86
5.5.4	Repository Breach by Drilling	5.87
5.5.5	Solution Mining	5.89
	REFERENCES FOR SECTION 5.5	5.93
5.6	COST OF GEOLOGIC DISPOSAL	5.94
5.6.1	Construction Costs	5.94
5.6.2	Operating Costs	5.96
5.6.3	Decommissioning Costs	5.96
5.6.4	Unit Costs	5.97
5.6.5	Comparison with Other Cost Data	5.98
5.6.6	Other Cost Considerations	5.98
	REFERENCES FOR SECTION 5.6	5.100
5.7	SAFEGUARDS INCLUDING PHYSICAL PROTECTION FOR GEOLOGICAL DISPOSAL	5.101
5.7.1	Geologic Disposal of Spent Fuel	5.101
5.7.2	Geologic Disposal of Solidified High-Level Waste and Transuranic Wastes	5.101
	REFERENCES FOR SECTION 5.7	5.102
5.8	IRREVERSIBLE AND IRRETRIEVABLE COMMITMENT OF RESOURCES ASSOCIATED WITH GEOLOGIC REPOSITORIES	5.103
5.9	SHORT-TERM USES OF THE ENVIRONMENT VERSUS LONG-TERM PRODUCTIVITY	5.104
5.10	UNAVOIDABLE ADVERSE ENVIRONMENTAL IMPACTS ASSOCIATED WITH RADIOACTIVE WASTE DISPOSAL IN GEOLOGIC REPOSITORIES	5.105
CHAPTER 6	- ALTERNATE CONCEPTS FOR WASTE DISPOSAL	6.1
6.1	PRESENTATION/ANALYSIS OF ALTERNATIVE CONCEPTS DISPOSAL	6.1
6.1.1	Very Deep Hole	6.6
6.1.1.1	Concept Summary	6.6
6.1.1.2	System and Facility Description	6.9
6.1.1.3	Status of Technical Development and R&D Needs	6.12
6.1.1.4	Impacts of Construction and Operation (Preplacement)	6.18
6.1.1.5	Potential Impacts Over the Long Term (Postplacement)	6.23
6.1.1.6	Cost Analysis	6.26
6.1.1.7	Safeguards	6.27

CONTENTS (contd)

6.1.2	Rock Melt	6.28
6.1.2.1	Concept Summary	6.28
6.1.2.2	System and Facility Description	6.28
6.1.2.3	Status of Technical Development and R&D Needs	6.35
6.1.2.4	Impacts of Construction and Operation (Preemplacement)	6.41
6.1.2.5	Potential Impacts Over the Long Term (Postemplacement)	6.46
6.1.2.6	Cost Analysis	6.47
6.1.2.7	Safeguard Requirements	6.47
6.1.3	Island Disposal	6.48
6.1.3.1	Concept Summary	6.48
6.1.3.2	System and Facility Description	6.48
6.1.3.3	Status of Technical Development and R&D Needs	6.54
6.1.3.4	Impacts of Construction and Operation (Preemplacement)	6.55
6.1.3.5	Potential Impacts Over Long Term (Postemplacement)	6.58
6.1.3.6	Cost Analysis	6.61
6.1.3.7	Safeguard Requirements	6.61
6.1.4	Subseabed	6.62
6.1.4.1	Concept Summary	6.62
6.1.4.2	System and Facility Description	6.63
6.1.4.3	Status of Technical Development and R&D Needs	6.68
6.1.4.4	Impacts of Construction and Operation (Preemplacement)	6.72
6.1.4.5	Potential Impacts Over Long Term (Postemplacement)	6.77
6.1.4.6	Cost Analysis	6.78
6.1.4.7	Safeguard Requirements	6.81
6.1.5	Ice Sheet Disposal	6.82
6.1.5.1	Concept Summary	6.82
6.1.5.2	System and Facility Description	6.82
6.1.5.3	Status of Technical Development and R&D Needs	6.89
6.1.5.4	Impacts of Construction and Operation (Preemplacement)	6.91
6.1.5.5	Potential Impacts Over Long Term (Postemplacement)	6.95

CONTENTS (contd)

6.1.5.6	Cost Analysis	6.97
6.1.5.7	Safeguard Requirements	6.97
6.1.6	Well Injection	6.100
6.1.6.1	Concept Summary	6.100
6.1.6.2	System and Facility Description	6.101
6.1.6.3	Status of Technical Development and R&D Needs	6.107
6.1.6.4	Impacts of Construction and Operation (Preemplacement)	6.112
6.1.6.5	Potential Impacts Over Long Term (Postemplacement)	6.115
6.1.6.6	Cost Analysis	6.117
6.1.6.7	Safeguard Requirements	6.118
6.1.7	Transmutation	6.119
6.1.7.1	Concept Summary	6.119
6.1.7.2	System and Facility Description	6.119
6.1.7.3	Status of Technical Development and R&D Needs	6.123
6.1.7.4	Impacts of Construction and Operation (Preemplacement)	6.126
6.1.7.5	Potential Impacts Over Long Term (Postemplacement)	6.132
6.1.7.6	Cost Analysis	6.133
6.1.7.7	Safeguard Requirements	6.135
6.1.8	Space Disposal	6.136
6.1.8.1	Concept Summary	6.136
6.1.8.2	System and Facility Description	6.136
6.1.8.3	Status of Technical Development and R&D Needs	6.142
6.1.8.4	Impacts of Construction and Operation (Preemplacement)	6.146
6.1.8.5	Potential Impacts Over Long Term (Postemplacement)	6.153
6.1.8.6	Cost Analysis	6.154
6.1.8.7	Safeguard Requirements	6.155
	REFERENCES FOR SECTION 6.1	6.156
6.2	COMPARISON OF ALTERNATIVE WASTE DISPOSAL CONCEPTS	6.165
6.2.1	Summary Description of Alternative Waste Disposal Concepts	6.165
6.2.1.1	Mined Repository	6.165

CONTENTS (contd)

6.2.1.2	Very Deep Hole	6.166
6.2.1.3	Rock Melting	6.167
6.2.1.4	Island Mined Repository	6.167
6.2.1.5	Subseabed Disposal	6.168
6.2.1.6	Ice Sheet Disposal	6.169
6.2.1.7	Well Injection	6.169
6.2.1.8	Transmutation	6.170
6.2.1.9	Space	6.171
6.2.1.10	Summary	6.171
6.2.2	Assessment Factors and Standards of Judgement	6.172
6.2.2.1	Radiological Effects	6.173
6.2.2.2	Status of Development	6.175
6.2.2.3	Conformance with Federal Law and International Agreements	6.175
6.2.2.4	Independence from Future Development of the Nuclear Industry	6.176
6.2.2.5	Cost of Development and Operation	6.176
6.2.2.6	Potential for Corrective Action	6.176
6.2.2.7	Long-Term Maintenance and Surveillance Requirements	6.177
6.2.2.8	Resource Consumption	6.177
6.2.2.9	Equity of Risk	6.177
6.2.3	Application of Performance Standards	6.178
6.2.3.1	A Concept Should Comply with Radiological Standards Established for Other Fuel Cycle Facilities	6.178
6.2.3.2	Containment Should be Maintained During the Period Dominated by Fission Product Decay	6.178
6.2.3.3	Waste Should Be Isolated from the Accessible Environment for a Minimum of 10,000 Years	6.179
6.2.3.4	The Concept Should be Amenable to Development Within a Reasonable Period of Time Such That Implementation is Not Left to Future Generations	6.181

CONTENTS (contd)

6.2.3.5	Implementation of a Concept Should Not Require Scientific Breakthroughs	6.181
6.2.3.6	Capabilities for Assessing the Performance of a Concept Must Be Available Prior to Committing Major R&D Programs to Its Development	6.181
6.2.3.7	Implementation of a Concept Should Not Be Dependent Upon the Size of the Nuclear Industry	6.182
6.2.3.8	Concepts Should Be Independent of Fuel Cycle Issues	6.182
6.2.3.9	Concepts Should Be Independent of Reactor Design Issues	6.183
6.2.3.10	Implementation of a Concept Should Allow Ability to Correct or Mitigate Failure	6.183
6.2.3.11	Maintenance or Surveillance Should Not Be Required for Extended Periods Following Termination of Active Repository Operations	6.183
6.2.3.12	Summary	6.184
6.2.4	Comparison of the Waste Disposal Concepts with Most Potential	6.184
6.2.4.1	Radiological Effects	6.184
6.2.4.2	Non-Radiological Environmental Effects	6.187
6.2.4.3	Status of Development	6.189
6.2.4.4	Conformance with Federal Law and International Agreements	6.191
6.2.4.5	Independence from Future Development of the Nuclear Industry	6.191
6.2.4.6	Cost of Development and Operation	6.191
6.2.4.7	Potential for Corrective or Mitigating Action	6.193
6.2.4.8	Long-Term Maintenance and Surveillance Requirements	6.194
6.2.4.9	Resource Consumption	6.194
6.2.4.10	Equity of Risk	6.194
6.2.5	Conclusions	6.194
6.2.5.1	Mined Repository	6.194
6.2.5.2	Subseabed	6.197
6.2.5.3	Very Deep Hole	6.197
6.2.5.4	Space Disposal	6.198

CONTENTS (contd)

6.2.5.5 Island Disposal	6.198
REFERENCES FOR SECTION 6.2	6.199
CHAPTER 7 - SYSTEM IMPACTS OF PROGRAM ALTERNATIVES	7.1
7.1 BASIS FOR SYSTEM SIMULATION	7.1
7.2 METHOD OF ANALYSIS FOR SYSTEM IMPACTS	7.6
7.3 SYSTEM LOGISTICS	7.11
7.3.1 Repository Inventory Accumulations	7.13
7.3.2 Interim Storage Requirements	7.18
7.3.3 Transportation Requirements	7.22
7.3.4 Age of the Waste at Disposal	7.24
7.3.5 Facility Requirements	7.26
7.3.6 Equilibrium Requirements for Equilibrium Steady-State Systems	7.29
7.3.7 Plutonium Disposition	7.30
7.3.8 Radioactivity Inventory in Disposal Repositories	7.32
7.4 SYSTEM RADIOLOGICAL IMPACTS	7.38
7.5 SYSTEM RESOURCE COMMITMENTS	7.42
7.6 SYSTEM COSTS	7.46
7.7 SYSTEM SIMULATION CONCLUSIONS	7.52
REFERENCES FOR CHAPTER 7	7.54
CHAPTER 8 - GLOSSARY OF KEY TERMS AND ACRONYMS	8.1

VOLUME 1FIGURES

1.1.1	Radioactivity in Spent Fuel and High-Level Waste as a Function of Time	1.4
1.1.2	Heat Generation Rate of Spent Fuel and High-Level Waste as a Function of Time	1.4
1.3.1	Deep Underground Geologic Waste Repository	1.8
2.1.1	Processes and Waste Streams in the Commercial Fuel Cycle	2.2
2.3.1	Site Characterization and Selection Process	2.11
3.2.1	Once-Through Cycle	3.9
3.2.2	Uranium-Plutonium Recycle Fuel Cycle	3.10
3.2.3	Nuclear Power Growth Assumptions	3.13
3.4.1	Toxicity of Spent Fuel and Reprocessing Waste from Uranium-Plutonium Recycle Relative to 0.2% Uranium Ore Necessary to Produce 1 MT of Reactor Fuel	3.38
4.1.1	Predisposal Waste Management System for Spent Fuel in the Once-Through Fuel Cycle	4.2
4.1.2	Predisposal Waste Management System for Fuel Reprocessing Plant and MOX-Fuel Fabrication Plant Wastes in the Fuel Reprocessing Cycle.	4.4
4.1.3	Example Predisposal Waste Management Operations for the Mined Geologic Disposal Option	4.8
4.2.1	Unirradiated Reference Fuel Assemblies	4.11
4.3.1	Flow Diagram for Encapsulation of Intact Spent Fuel Assemblies	4.18
4.3.2	Flow Diagram for Encapsulation of Chopped and Voloxidized Spent Fuel	4.19
4.3.3	Flow Diagram for Encapsulation of Dissolved and Vitrified Spent Fuel	4.21
4.3.4	Flow Diagram for Dissolution of Spent Fuel for Disposal as a Liquid	4.23
4.3.5	Flow Diagram for Spray Calciner/In-Can Melting Process	4.24
4.3.6	Flow Diagram for Fuel Residue Packaging Without Compaction	4.33
4.3.7	Flow Diagram of Mechanical Compaction of Hulls	4.34
4.3.8	Flow Diagram for Treatment of Failed Equipment and Noncombustible Waste at an FRP	4.35
4.3.9	Treatment of Combustible Wastes and Filters at FRP Remotely Handled Waste Incinerator Facility	4.37

FIGURES (contd)

4.3.10	Process Flow Diagram for Cementation at Fuel Reprocessing Plant	4.38
4.3.11	Process Flow Diagram for Bitumenization Facility at Fuel Reprocessing Plant	4.40
4.3.12	Flow Diagram for Filtration of Airborne Wastes	4.41
4.3.13	Flow Diagram for Gaseous Radionuclide Recovery	4.42
5.1.1	Deep Underground Geologic Waste Repository	5.2
5.1.2	Conceptual Waste Package	5.7
5.3.1	Plot Plan of a Geologic Repository	5.32
5.3.2	Artist's Concept of a Geologic Repository and Its Support Facilities	5.33
5.3.3	Effect of Spent Fuel Age on Once-Through Cycle Repository Capacities	5.43
5.3.4	Effect of HLW Age on Reprocessing Cycle Repository Capacities	5.43
6.1.1	Major Options for Very Deep Hole Disposal of Nuclear Waste	6.7
6.1.2	Waste Management System--VDH Disposal	6.8
6.1.3	Major Options for Rock Melting Disposal Of Nuclear Waste	6.29
6.1.4	Waste Management System-Rock Melting Disposal	6.31
6.1.5	Schematic Illustration of Hydrous and Anhydrous Melting Intervals for an Average Granite	6.32
6.1.6	Radius of Waste-Rock Melt Pool Over Time (for typical Cavity and Waste Loading)	6.32
6.1.7	Major Options for Island Disposal of Nuclear Waste	6.50
6.1.8	Waste Management System--Island Disposal	6.51
6.1.9	Hydrological Classification of Repository Locations	6.52
6.1.10	Isolation Barriers for Fresh Water Lens Location	6.60
6.1.11	Isolation Barriers for Saline Zone Location	6.60
6.1.12	Major Options for the Subseabed Disposal of Nuclear Waste	6.64
6.1.13	Waste Management System--Subseabed Disposal	6.65
6.1.14	Major Option for Ice Sheet Disposal of Nuclear Waste	6.83
6.1.15	Waste Management System-Ice Sheet Disposal	6.85
6.1.16	Schematic of Operations in Ice Sheet Disposal Systems for High-Level Radioactive Wastes ⁽¹⁸⁾	6.86

FIGURES (contd)

6.1.17	Ice Sheet Emplacement Concepts	6.87
6.1.18	Major Options for Well Injection Disposal of Nuclear Waste	6.101
6.1.19	Waste Management System--Well Injection Disposal	6.103
6.1.20	Major Options for a Waste Disposal Alternative Using Transmutation	6.120
6.1.21	Partitioning-Transmutation Fuel Cycle Diagram	6.121
6.1.22	Major Options for Space Disposal of Nuclear Waste	6.137
6.1.23	Waste Management System--Space Disposal	6.139
6.1.24	Orbital Operations	6.141
7.2.1	System Simulation Information Flow	7.6
7.2.2	WASTRAC Calculations	7.8
7.2.3	Time and Discounting Relationships of Waste Management Functions of Cost	7.9
7.3.1	Repository Inventory Accumulations for Cases 1 and 2.	7.13
7.3.2	Repository Inventory Accumulation for the Once-Through Cycle in Case 3	7.14
7.3.3	Cumulative Fuel Reprocessed for Case 3	7.15
7.3.4	Repository High-Level Waste Inventory Accumulation for Case 3 with Reprocessing	7.15
7.3.5	Repository Inventory Accumulation for The Once-Through Cycle in Cases 4 and 5	7.16
7.3.6	Cumulative Fuel Reprocessed for Cases 4 and 5	7.16
7.3.7	Repository High-Level Waste Inventory Accumulation for Cases 4 and 5 with Reprocessing	7.17
7.3.8	AFR Storage Requirements for Case 3 with the Once-Through Cycle	7.20
7.3.9	AFR Storage Requirements for Case 3 with Reprocessing	7.20
7.3.10	Age of Fuel Entering Repository for Case 3 with the Once-Through Cycle	7.25
7.3.11	Age of High-Level Waste Entering Repository for Case 3 with the Reprocessing Cycle	7.26

VOLUME 1TABLES

1.1.1	Total Spent Fuel Disposal or Reprocessing Requirements	1.3
1.3.1	Number of 800 Hectare Example Repositories Required for Various Nuclear Power Growth Assumptions	1.9
1.3.2	Resource Commitments Associated with Construction and Operation of Geologic Waste Repositories, Normalized to 1 GWe-yr	1.11
1.3.3	Manpower Requirements for Construction and Operation of a Single Waste Repository (three peak years)	1.12
1.6.1	Summary of Radiation Effects from Potential Worst-Case Predisposal System Accidents	1.23
1.7.1	Repository Startup Dates Considered in the Once-Through-Cycle System Simulations	1.25
1.7.2	Reprocessing and Repository Startup Date Combinations Considered in the Reprocessing-Cycle System Simulations	1.25
1.7.3	Comparison of 70-Year Whole-Body Dose Accumulations from Normal Operations for the Program Alternatives Using the Once-Through Cycle, man-rem	1.26
1.7.4	Comparison of 70-Year Whole-Body Dose Accumulations from Normal Operations for the Program Alternatives Using the Reprocessing Cycle, man-rem	1.27
1.7.5	Comparison of Health Effects for the Program Alternatives Using the Once-Through Cycle	1.28
1.7.6	Comparison of Health Effects for the Program Alternatives Using the Reprocessing Cycle	1.28
1.7.7	Comparison of Levelized Waste-Management Unit Costs for the Program Alternatives at a 7% Discount Rate, ¢/kWh	1.30
3.1.1	Potential Ability of Technology to Handle Waste Type	3.2
3.2.1	Nuclear Power Capacity Assumptions, GWe	3.12
3.3.1	Summary of Average Whole-Body Dose-Equivalent Rates from Naturally Occurring Radiation, mrem/yr	3.29
3.3.2	Nominal Whole-Body Dose Equivalents from Naturally Occurring Radiation	3.30
3.3.3	Health Effects Calculated for 70-yr Accumulated Dose from Naturally Occurring Radioactive Sources	3.30
3.4.1	The Relative Toxicity (Hazard) of Various Ores Compared to U Ore (0.2%)	3.37
3.4.2	Potential Disruptive Phenomena for Waste Isolation Repositories	3.40
4.1.1	Predisposal Operations and Alternatives for Once-Through Cycle Disposal Options	4.3

TABLES (contd)

4.1.2	Predisposal Operations and Alternatives for Reprocessing-Cycle High-Level Liquid Wastes	4.5
4.1.3	Example Predisposal Operations and Alternatives Evaluated for Reprocessing-Cycle TRU Wastes for All Disposal Concepts	4.6
4.2.1	Classification of Primary Wastes from the Post-Fission LWR Fuel Cycle	4.10
4.2.2	Selected Radionuclide Content in Example Once-Through Cycle Spent Fuel	4.12
4.2.3	Selected Radionuclide Content in Primary High-Level, TRU, and Gaseous Wastes from Fuel Reprocessing Plant and MOX Fuel Fabrication Plant	4.14
4.2.4	Selected Radionuclide Content in Example Recycle Spent Fuel	4.15
4.2.5	Selected Radionuclide Content in Example MOX fuel	4.16
4.3.1	Estimated Radionuclide Releases During Waste Treatment and Packaging	4.45
4.3.2	Estimated Quantities of Packaged High-Level, TRU, and Gaseous Wastes	4.46
4.4.1	Estimated Radionuclide Releases During Water Basin Storage of Unpackaged Spent Fuel	4.53
4.4.2	Estimated Radionuclide Releases During Tank Storage of Liquid High-Level Waste	4.56
4.4.3	Estimated Radionuclide Releases During Water Basin Storage of Vitrified High-Level Waste	4.57
4.5.1	Available Shipping Casks for Current Generation LWR Spent Fuel	4.63
4.6.1	Volumes and Radionuclide Content of TRU Wastes Resulting from Decommissioning of Reprocessing Cycle Facilities	4.72
4.6.2	Estimated Quantities of Packaged TRU-Decommissioning Wastes	4.73
4.6.3	Radionuclides Released on Example Decommissioning of Facilities	4.74
4.7.1	Resource Commitments for Construction and Operation of an Example AFR	4.77
4.7.2	Resource Commitments for Fabrication and Use of Spent Fuel Shipping Casks	4.77
4.7.3	Doses Resulting From Operation and Decommissioning of an AFR	4.78
4.7.4	Selected Social Service Demands Associated with In-Migration Related to a 3000 MTHM AFR	4.80
4.7.5	Resource Commitments for Construction and Operation of Reprocessing Fuel Cycle Waste Management Facilities	4.82
4.7.6	Resource Commitments for Construction and Use of Waste Shipping Containers	4.83

TABLES (contd)

4.7.7	Dose to Regional Population Due to Operation of an FRP and a MOX-FFP	4.85
4.7.8	Example Reprocessing Cycle Waste Management Operations at Individual Facilities	4.86
4.7.9	Selected Social Service Demands Associated with In-Migration Related to Waste Management Facilities at an FRP, a MOX-FFP, and an RWSF	4.88
4.8.1	Disabling Injuries and Deaths from Construction and Decommissioning of Reprocessing Fuel Cycle Waste Management Facilities	4.98
4.8.2	Summary of Radiation Effects from Potential Worst-Case Predisposal System Accidents	4.99
4.9.1	Unit Costs of Predisposal Operations for Once-Through Cycle Disposal Options	4.102
4.9.2	Unit Costs of Predisposal Operations for Reprocessing Waste Disposal Operations	4.104
4.9.3	Predisposal Unit Costs for the Once-Through Cycle	4.106
4.9.4	Unit Cost Estimates for Reprocessing Fuel Cycle Wastes	4.107
4.9.5	Unit Cost Estimates for Interim Storage Operations for Reprocessing Fuel Cycle Wastes	4.109
4.9.6	Unit Cost Estimates for Example Transportation Operations, \$/kg HM	4.109
4.9.7	Subsystems Waste Management Costs for Alternative Waste Treatment Options	4.110
5.3.1	Conceptual Repository Design Thermal Limits for Spent Fuel	5.30
5.3.2	Mining and Rock Handling Requirements at the Reference Spent Fuel Repository	5.34
5.3.3	Contents of the Conceptual Spent Fuel Repositories When Full	5.35
5.3.4	Conceptual Repository Design Thermal Limits for Reprocessing Cycle Wastes	5.38
5.3.5	Conceptual Repository Thermal Limits for Individual HLW Waste Canisters	5.38
5.3.6	Mining and Rock Handling Requirements at the Reference Reprocessing Waste Repository	5.39
5.3.7	Contents of the Conceptual Reprocessing Waste Repositories When Full	5.41
5.4.1	Land Use Commitments For Construction of 800-ha Single Geologic Repositories	5.46

TABLES (contd)

5.4.2	Resource Commitments Necessary for Construction of a Spent Fuel Repository in Salt, Granite, Shale, and Basalt	5.47
5.4.3	Resource Commitments Necessary for Construction of a Fuel Reprocessing Waste Repository in Salt, Granite, Shale, and Basalt	5.47
5.4.4	Quantities of Effluents Released to the Atmosphere During Construction of a Geologic Repository	5.48
5.4.5	Maximum Dust Emissions From Surface Handling of Mined Material, MT/d	5.49
5.4.6	Particulate Concentrations at Repository Fenceline, g/m ³	5.50
5.4.7	Dust Depositions from Surface Handling of Mined Material, gm/m ² -yr	5.50
5.4.8	Annual Releases of Naturally Occurring Radionuclides to Air For Construction of Geologic Repository for Spent Fuel, Ci	5.52
5.4.9	Annual Releases of Naturally Occurring Radionuclides to Air for Construction of Geologic Repository for Fuel Reprocessing Waste, Ci	5.52
5.4.10	Summary of 70-Yr Whole-Body Dose Commitments from Naturally Occurring Radioactive Sources During Mining Operations at a Repository, Man-rem	5.53
5.4.11	Estimates of Nonradiological Disabling Injuries and Fatalities Associated with Repository Construction Based Upon Current Industrial Statistics for Similar Operations	5.57
5.4.12	Resource Commitments for the Operational Phase of Spent Fuel Geologic Repositories	5.58
5.4.13	Resource Commitments for the Operational Phase of Fuel Reprocessing Waste Geologic Repositories	5.59
5.4.14	Total Quantities of Effluents Released to the Atmosphere During Operation of a Geologic Repository for Spent Fuel	5.59
5.4.15	Total Quantities of Effluents Released to the Atmosphere During Operation of Geologic Repository for Reprocessing Wastes	5.60
5.4.16	Estimated Manpower Requirements for Construction and Operation of a Single Waste Repository, by Disposal Average Annual Employment, (3-yr Peak)	5.62
5.4.17	Forecasts of Expected Population Influx for a Geologic Repository in Salt (51,000 MTHM Waste Capacity): Number of Persons and Percent of Base Population	5.62
5.4.18	Selected Expected Social Service Demands Associated with Migration into the Site County Resulting from the Construction and Operation of a Geologic Repository in Salt	5.64
5.4.19	Selected Expected Social Service Demands Associated with Migration into the Site County Resulting from the Construction and Operation of a Geologic Repository in Granite	5.64

TABLES (contd)

5.4.20	Selected Expected Social Service Demands Associated with Migration into the Site County Resulting from the Construction and Operation of a Geologic Repository in Shale	5.65
5.4.21	Selected Expected Social Service Demands Associated with Migration into the Site County Resulting from the Construction and Operation of a Geologic Repository in Basalt	5.65
5.4.22	Radioactive Material Released to the Atmosphere from a Spent Fuel Canister Drop-Down-Mine-Shaft Accident at a Geologic Repository	5.67
5.4.23	Postulated Accidents for the Geologic Repository for Reprocessing Wastes	5.67
5.4.24	Radioactive Material Released to the Atmosphere from a CH-TRU Waste Accident at the Geologic Repository for Reprocessing Wastes, Ci	5.67
5.4.25	Radionuclide Releases for a Waste Canister Dropped Down a Mine Shaft at a Repository for Reprocessing Wastes, Ci	5.68
5.4.26	70-Yr Whole-Body Dose Commitments to Maximum Individual from Drop of Waste Canisters into a Geologic Repository	5.69
5.5.1	First-Year Whole-Body Dose to Maximum Individual--Repository Breach by Meteorite Strike, rem	5.76
5.5.2	70-Year Whole-Body Dose Commitment to Maximum Individual--Repository Breach by Meteorite Strike, rem	5.76
5.5.3	70-Year Whole Body-Dose Commitment to the Regional Population--Repository Breach by Meteorite, man-rem	5.77
5.5.4	70-Year Cumulative Whole-Body Dose to First Five Generations of Regional Population--Repository Breach by Meteorite, man-rem	5.78
5.5.5	70-Year Whole-Body Dose Commitment to Population of Eastern United States--Repository Breach by Meteorite, Strike, man-rem	5.79
5.5.6	Estimated Leach Rates for Various Forms of Radioactive Wastes Used in Consequence Analyses	5.81
5.5.7	70-Year Whole-Body Dose Commitment to Maximum Individual--Repository Breach by Faulting and Flooding, rem	5.82
5.5.8	70-Year Whole-Body Dose Commitment to the Regional Population--Repository Breach by Faulting and Flooding	5.83
5.5.9	70-Yr Accumulated Whole-Body Dose to Maximum Individual for Various Leach Rates and Times of Repository Breach by Fracturing and Ground-Water Intrusion (repository in salt--50,000 MTHM), rem	5.85
5.5.10	Respirable Radionuclides Released to the Atmosphere from Salt Repository Breach by Drilling 1000 Yrs After Repository Closure, Ci	5.88
5.5.11	Amounts of Radionuclides (Ci) and 70-Year Whole-Body Dose (in rems to an individual) Resulting from Ingestion of 1800 g of Contaminated Salt 1000 years after Repository Closure	5.91

TABLES (contd)

5.6.1	Cost Estimates for 800-hectare Geologic Repositories	5.95
5.6.2	Decommissioning Costs for Spent Fuel and Reprocessing-Waste Repositories	5.97
5.6.3	Unit Costs by Waste Type and Geologic Media	5.97
5.6.4	Spent Fuel Retrieval Costs	5.98
6.1.1	Radiological Impact-Routine Operation (Bechtel 1979a)	6.19
6.1.2	VDH Concept-Occupational Doses During Normal Operation (Bechtel 1979a)	6.20
6.1.3	Radiological Impact-Abnormal Conditions	6.20
6.1.4	Nonradiological Environmental Impact	6.22
6.1.5	Estimated Energy Consumption	6.23
6.1.6	Estimated Consumption of Critical Resources	6.24
6.1.7	Long-Term Radiological Impact of Primary Waste Barrier Breach	6.26
6.1.8	Occupational Dose Estimate During Normal Operation At a Single Rock Melting Cavity	6.43
6.1.9	Estimated Energy Consumption (Bechtel 1979a)	6.45
6.1.10	Estimated Material Consumption (Metric Tons)	6.45
6.1.11	Radiological Impacts of the Normal Operation At a Subseabed Repository	6.73
6.1.12	Estimated Dose Commitment From Marine Food Chain For Loss of Waste at Sea	6.74
6.1.13	Estimated Energy Consumption	6.76
6.1.14	Estimated Material Consumption for Ship and Facility Construction (in MT)	6.76
6.1.15	Levels of Natural and Wastes Radionuclides in Seawater	6.78
6.1.16	Estimating Operating Costs	6.80
6.1.17	Capital Costs for Ice Sheet Disposal (Millions of 1978 Dollars)	6.98
6.1.18	Operating Costs For Ice Disposal (Millions of 1978 Dollars/Years)	6.99
6.1.19	Reference Concepts Summary (DOE 1979)	6.102

TABLES (contd)

6.1.20	Estimated Transmutation R&D Costs and Implementation Time	6.125
6.1.21	Annual Routine Radiological Occupational Dose	6.127
6.1.22	Annual Routine Non-Occupational Dose	6.127
6.1.23	Occupational Radiological Exposure--Abnormal	6.128
6.1.24	Non-Occupational Radiological Exposure--Abnormal	6.128
6.1.25	Transportation Non-Occupational Radiological Exposures--Abnormal	6.129
6.1.26	Summary Effects (Per Plant-Year) of Non-Radiological Effluents (Fullwood and Jackson 1980)	6.130
6.1.27	Capital Costs for Partitioning Facilities (Million of 1978 Dollars) (Smith and Davis 1978)	6.134
6.1.28	Operating Costs for Partitioning Facilities (Millions of 1980 Dollars)	6.135
6.1.30	Short Term (Preplacement Radiological Impacts For the Space Disposal Program Normal Operation	6.148
6.2.1	Disposition of Principal Waste Products Using the Proposed Waste Disposal Concepts	6.166
6.2.2	Assessment Factors	6.172
6.2.3	Standards of Judgement	6.173
6.2.4	Proposed DOE Waste Management Performance Objectives	6.174
6.2.5	Potentially Critical Materials	6.177
6.2.6	Performance of Proposed Waste Management Concepts on Ten Performance Standards	6.185
6.2.7	Estimated costs of Various Disposal Options (1980 dollars)	6.192
6.2.8	Estimated Resource Commitments for Various Repositories	6.195
6.2.9	Summary of Preference Rankings	6.196
7.1.1	Electric Energy Generated in Nuclear Power Growth Scenarios	7.2
7.1.2	Repository Startup Dates Considered in the Once-Through-Cycle System Simulations	7.4
7.1.3	Reprocessing and Repository Startup Date Combinations Considered in the Reprocessing-Cycle System Simulations	7.4
7.3.1	Total Spent Fuel Disposal or Reprocessing Requirements	7.13

TABLES (contd)

7.3.1a	Number of Spent Fuel Canisters Sent to Disposal in the Once-Through Cycle	7.17
7.3.1b	Number of Waste Containers Sent to Disposal in Reprocessing Cycle	7.18
7.3.2	Comparison of Away-From-Reactor Spent Fuel Storage Requirements for the Program Alternative Using the Once-Through Cycle	7.19
7.3.3	Comparison of Away-From-Reactor Spent Fuel Storage Requirements for the Program Alternative Using the Reprocessing Cycle	7.19
7.3.4	Interim Waste Storage Requirements for the Program Alternatives Using the Reprocessing Cycle	7.21
7.3.5	Comparison of Transportation Requirements for the Program Alternative Using the Once-Through Fuel Cycle	7.22
7.3.6	Comparison of Total Transportation Requirements for the Program Alternative Using the Reprocessing Fuel Cycle	7.23
7.3.7	Maximum (and Minimum) Age of Spent Fuel Entering the Repository Using the Once-Through Cycle, Years	7.24
7.3.8	Maximum (and minimum) Age of High-Level Waste Entering the Repository using the Reprocessing Cycle, Years	7.25
7.3.9	Fuel Reprocessing and MOX Fuel Fabrication Plant Requirements	7.28
7.3.10	Number of 800-hectare Repositories Required	7.29
7.3.11	Equilibrium Requirements for Case 4 (250 GWe Steady State)	7.29
7.3.12	Plutonium Disposition Within the Timeframe of the Analysis	7.31
7.3.13	Total Radioactivity Inventory of All Fission and Activation Products in All Repositories	7.33
7.3.14	Total Radioactivity Inventory of All Actinide and Daughter Nuclides in All Repositories	7.33
7.3.15	Heat Output of Total Inventory of all Fission and Activation Products in All Repositories	7.34
7.3.16	Heat Output of Total Inventory of All Actinide and Daughter Nuclides in All Repositories	7.34
7.3.17	Hazard Index of Repository Waste Inventory Relative to 0.2% Uranium Ore.	7.36
7.3.18	Principal Contributors to the Hazard Index	7.37
7.4.1	Comparison of 70-Year Whole-Body Dose Accumulations from Normal Operations for the Program Alternatives Using the Once-Through Cycle, man-rem	7.38

TABLES (contd)

7.4.2	Comparison of 70-Year Whole-Body Dose Accumulations from Normal Operations for the Program Alternatives Using the Reprocessing Cycle, man-rem	7.39
7.4.3	Comparison of Normal Operations Health Effects for the Program Alternatives Using the Once-Through Cycle (number of deaths and/or genetic defects)	7.40
7.4.4	Comparison of Normal Operations Health Effects for the Program Alternatives Using the Reprocessing Cycle (number of deaths and/or genetic defects)	7.40
7.5.1	Resource Commitment Reference Cases	7.43
7.5.2	Comparison of Relative Resource Commitments for the Program Alternatives Using the Once-Through Fuel Cycle	7.44
7.5.3	Comparison of Relative Resource Commitments for the Program Alternatives Using the Reprocessing Cycle	7.45
7.6.1	Total Estimated Research and Development and Multiple Site Qualification Costs, \$ millions	7.46
7.6.2	Comparison of Total Waste Management Costs for the Program Alternatives Using the Once-Through Cycle, \$ Billions	7.47
7.6.3	Comparison of Total Waste Management Costs for the Program Alternatives Using the Reprocessing Cycle, \$ Billions	7.48
7.6.4	Comparison of Levelized Waste-Management Unit Costs for the Program Alternatives Using the Once-Through Cycle and a 0% Discount Rate, mills/kWh	7.49
7.6.5	Comparison of Levelized Waste-Management Unit Costs for the Program Alternatives Using the Reprocessing Cycle and a 0% Discount Rate, mills/kWh	7.49
7.6.6	Comparison of Levelized Waste-Management Unit Costs for the Program Alternatives Using the Once-Through Cycle and a 7% Discount Rate, mills/kWh	7.50
7.6.7	Comparison of Levelized Waste-Management Unit Costs for the Program Alternatives Using the Reprocessing Cycle and a 7% Discount Rate, mills/kWh	7.50
7.6.8	Comparison of Levelized Waste-Management Costs for the Program Alternatives Using the Once-Through Cycle and a 10% Discount Rate, mills/kWh	7.51
7.6.9	Comparison of Levelized Waste-Management Costs for the Program Alternatives Using the Reprocessing Cycle and a 10% Discount Rate, mills/kWh	7.51

CHAPTER 1

SUMMARY

In the course of producing electrical power in light water reactors (LWRs), the uranium fuel accumulates fission products until the fission process is no longer efficient for power production. At that point the fuel is removed from the reactor and stored in water basins to allow radioactivity to partially decay before further disposition. This fuel is referred to as "spent fuel." Although spent fuel as it is discharged from a reactor is intensely radioactive, it has been stored safely in moderate quantities for decades. Spent fuel could be reprocessed, and about 99.5% of the remaining uranium and newly formed plutonium could be recovered for reuse. However, present policy dictates that spent LWR fuel reprocessing will be indefinitely deferred because of concern that widespread separation of plutonium could lead to proliferation of nuclear weapons. As a result, spent fuel is currently stored for possible future reprocessing or disposal. Storage or disposal must be designed so that nuclear waste will not be a present or future threat to public health and safety.

The United States Department of Energy (DOE) has the responsibility to develop technologies for management and disposal of certain classes of commercially generated radioactive wastes (namely high-level and transuranic).^(a) High-level waste is defined as either the aqueous solution from the first-cycle solvent extraction, where spent fuel is reprocessed for recycle of uranium and plutonium, or spent fuel if disposed of. High-level waste is also intensely radioactive.

Other wastes are generated during reprocessing that, although larger in volume than high-level wastes, are less intensely radioactive. Wastes that contain more than a specified amount of radionuclides of atomic number greater than that of uranium are called transuranic (TRU) wastes. TRU wastes are categorized here as either remotely handled (RH) or contact-handled (CH) wastes, depending on the requirements for radiation protection of personnel. Special attention must be given to TRU wastes because they contain alpha particle-emitting nuclides that are of particular concern as a result of their long half lives and tenacious retention if incorporated in the body. Other waste forms that include neither high-level nor TRU are so-called low-level wastes.^(b)

The principal objective of waste disposal is to provide reasonable assurance that these wastes, in biologically significant concentrations, will be permanently isolated from the human environment. To provide input to the decision on a planning strategy for disposal of these radioactive wastes, this Statement presents an analysis of environmental impacts that could occur if various technologies for management and disposal of such wastes were to be developed and implemented.

(a) In a message to Congress on February 12, 1980, the President reiterated the role of DOE as lead agency for management and disposal of radioactive wastes.

(b) Low level wastes, other than those originating at DOE facilities, are managed and disposed of by licenses in accordance with regulations of the NRC.

The DOE is proposing a program strategy emphasizing development of conventionally mined waste repositories, deep in the earth's geologic formations, as a means of disposing of commercially-generated high-level and TRU wastes. Adoption of this program strategy constitutes a major federal action for which the National Environmental Policy Act of 1969 (NEPA) requires preparation of a detailed environmental impact statement (EIS).

This summary highlights the major findings and conclusions of this final Statement. It reflects the public review of and comments offered on the draft Statement. Included are descriptions of the characteristics of nuclear waste, the alternative disposal methods under consideration, and potential environmental impacts and costs of implementing these methods. Because of the programmatic nature of this document and the preliminary nature of certain design elements assumed in assessing the environmental consequences of the various alternatives, this study has been based on generic, rather than specific, systems. At such time as specific facilities are identified for particular sites, statements addressing site-specific aspects will be prepared for public review and comment.

1.1 THE NEED FOR WASTE MANAGEMENT AND DISPOSAL

There are now about 70 operating commercial LWR power reactors in the United States, which represent approximately 50 GWe^(a) of installed nuclear powered electrical generating capacity. The amounts of spent fuel accumulated for the present (1980) inventory and for alternative nuclear power generating scenarios considered in this Statement are shown in Table 1.1.1.

TABLE 1.1.1. Total Spent Fuel Disposal or Reprocessing Requirements

Case	Scenario	Nuclear Power Growth Assumption	
		Energy Generated, GWe-yr. ^(a)	Spent Fuel Discharged, MTHM ^(b)
1	Present Inventory Only-- Reactors Shut Down in 1980 ^(c)	200	10,000
2	Present Capacity (50 GWe) ^(c) and Normal Reactor Life	1,300	48,000
3	250 GWe System by Year 2000 and Normal Reactor Life (No new reactors after Year 2000) ^(d)	6,400	239,000
4	250 GWe System by Year 2000 and Steady State Capacity to Year 2040 (New reactors to maintain output) ^(d)	8,700	316,000
5	500 GWe System by Year 2040 ^(d)	12,100	427,000

- (a) Energy generated is based on the total accumulated through the year 2040.
- (b) MTHM = metric tons (1000 kg = about 1.1 U.S. tons) of heavy metal in original fuel. One MTHM of spent fuel consists of about 96% uranium, 1% plutonium and 3% fission products.
- (c) Reprocessing is not applicable to Cases 1 and 2 because in Case 1 there is no need for reprocessing and in Case 2 no economic incentives exist for reprocessing.
- (d) Waste management impacts of nuclear power generation through the year 2040 are considered for these scenarios.

The total radioactivity in one MTHM of LWR fuel and equivalent HLW for various times after discharge from a reactor is shown in Figure 1.1.1. Similarly, the heat generation rate in this fuel is illustrated in Figure 1.1.2. These figures show that a reduction by a factor of about 1,000 in radioactivity relative to one-year-old fuel is reached in about 700 years for spent fuel and in about 200 years for uranium and plutonium recycle high-level waste. The heat generation rate is lower by a factor of 100 for spent fuel at about 300 years and for recycle high-level waste at about 150 years.

(a) One GWe = 1×10^9 watts.

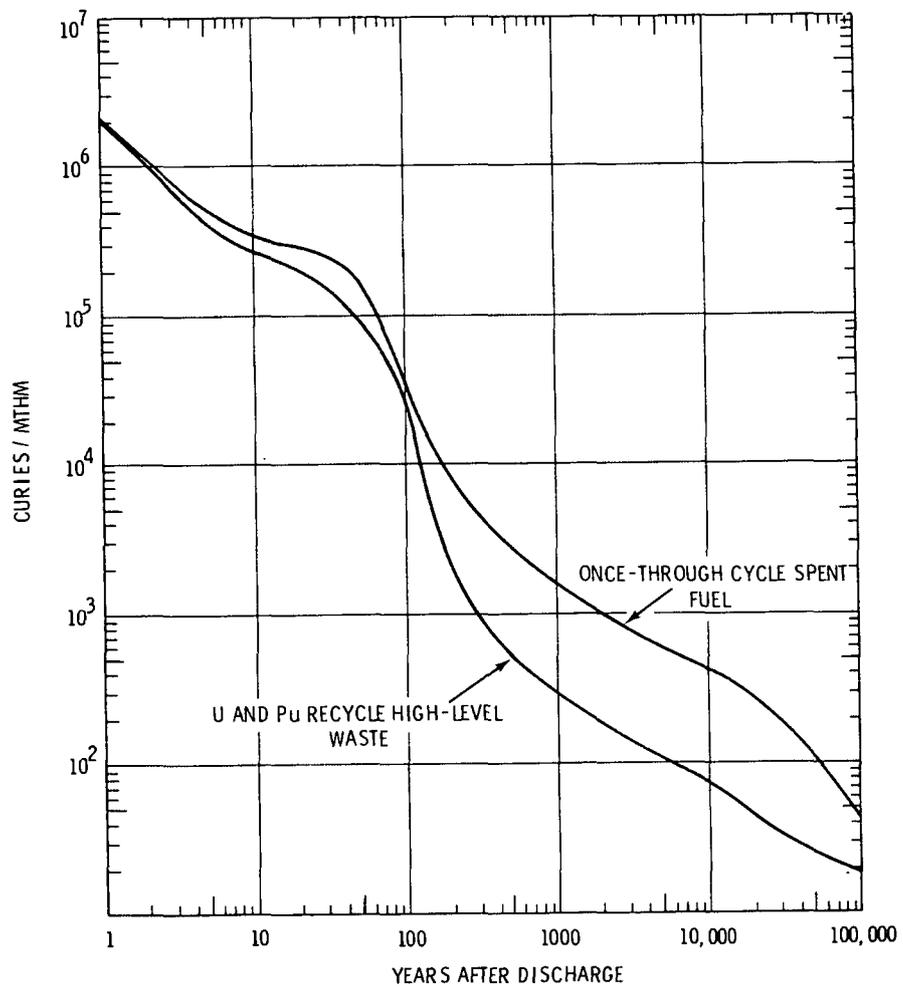


FIGURE 1.1.1. Radioactivity in Spent Fuel and High-Level Waste as a Function of Time

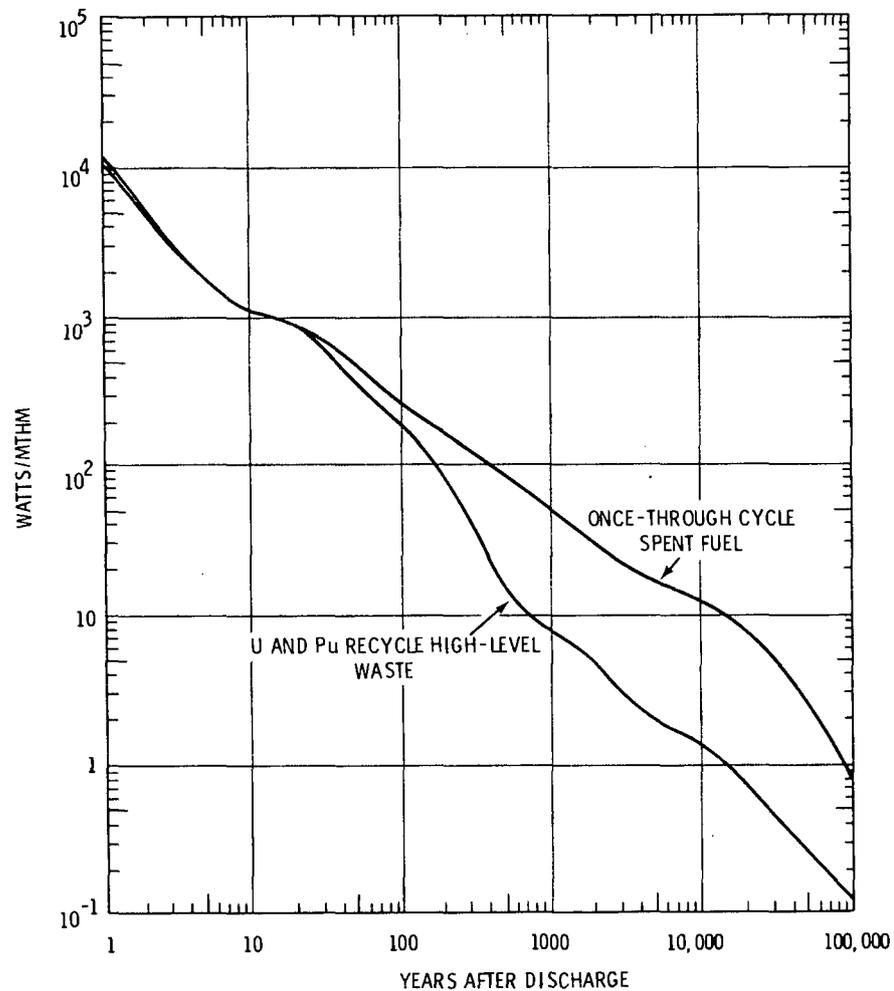


FIGURE 1.1.2. Heat Generation Rate of Spent Fuel and High-Level Waste as a Function of Time

The President, in his February 12, 1980 message on radioactive wastes, called for waste disposal facilities that could receive wastes from both the commercial nuclear power production program and the national defense program. Since defense wastes are not explicitly treated in this Statement, it is not intended to provide environmental input for disposal decisions on defense wastes. However, in a generic sense, systems that can adequately dispose of commercial radioactive wastes can reasonably be expected to adequately dispose of defense wastes, since the processed wastes from the national defense program produce lower temperatures and lower radiation intensities than do wastes from the same quantity of similarly processed commercial fuel. Thus, assuming that other factors are equal, repository loading criteria would generally be less stringent (in terms of quantities of waste per unit area) for defense wastes than for commercial wastes. For this reason certain of the analyses of impacts presented in this EIS should be of use in the preparation of EIS's on the long term management of high-level and TRU defense waste.

1.2 THE PROGRAMMATIC ALTERNATIVES

The programmatic alternatives considered in this Statement are:

- Proposed Action. The research and development program for waste management will emphasize use of mined repositories in geologic formations in the continental U.S. capable of accepting radioactive wastes from either the once-through or reprocessing cycles (while continuing to examine subseabed and very deep hole disposal as potential backup technologies). This action will be carried forward to identify specific locations for the construction of mined repositories. The proposed action does not preclude further study of other disposal techniques. For example, the selective use of space disposal for specific isotopes might be considered.
- Alternative Action. The research and development program would emphasize the parallel development of several disposal technologies. This action implies an R&D program to bring the knowledge regarding two or three disposal concepts and their development status to an approximately equal level. Based upon the Department of Energy's current evaluation, the likely candidate technologies for this parallel development strategy would be:
 - 1) geologic disposal using conventional mining techniques
 - 2) placement in sediment beneath the deep ocean (subseabed)
 - 3) disposal in very deep holes.

At some later point, a preferred technology would be selected for construction of facilities for radioactive waste disposal.

- No Action Alternative. This alternative would eliminate or significantly reduce the Department of Energy's research and development programs for radioactive waste disposal. Under this alternative, existing spent fuel would be left indefinitely where it is currently stored and any additional spent fuel discharged from future operation of commercial nuclear power plants would likewise be stored indefinitely in water basin facilities either at the reactors or at independent sites.

1.3 THE PROPOSED ACTION

The proposed action is to select and pursue a programmatic strategy that would lead to disposal of existing and future commercially generated radioactive high-level and transuranic wastes in mined repositories in geologic formations. This Statement addresses environmental impacts related to implementing such disposal^(a). The programmatic strategy will direct effort and concentrate resources on a research and development program leading to repositories and to site-selection processes. Some support will be provided to further evaluate the alternatives of subseabed disposal and disposal in very deep holes.

Environmental impacts related to repository construction, operation, and decommissioning are analyzed in this Statement as are the impacts of predisposal waste treatment, storage and transportation to the extent they might effect selection of a disposal option. Environmental impacts are developed for individual example facilities and for systems based on the power growth scenarios described in Table 1.1.1. This very broad or generic approach to evaluating the environmental issues provides a comprehensive overview of the likely consequences of the proposed action and constitutes the first phase of DOE's NEPA implementation plan for waste management and disposal (DOE/NE-0007 1980). This plan for waste management and disposal is based on a tiered approach, which is designed to eliminate repetitive discussions on the same issues and to focus on important issues ready for decision at each level of environmental review. Thus, as more site- or facility-specific decision points are approached, and before each such decision and before conducting of activities that may cause an adverse impact or limit the choice of reasonable alternatives, additional environmental assessments, or impact statements will be prepared as appropriate.

The proposed research and development program for waste management will emphasize use of mined repositories in geologic formations capable of accepting radioactive wastes from either the once-through or reprocessing cycles. This program will be carried forward to identify specific locations for the construction of mined repositories.

Initially, site characterization programs will be conducted to identify qualified sites in a variety of potential host rock and geohydrologic settings. As qualified sites are identified by the R&D program, actions will be taken to reserve the option to use the sites, if necessary, at an appropriate time in the future. Supporting this site characterization and qualification program will be research and development efforts to produce techniques and equipment to support the placement of wastes in mined geologic repositories.

The Department of Energy proposes that the development of geologic repositories will proceed in a careful step-by-step fashion. Experience and information gained in each phase of the development program will be reviewed and evaluated to determine if there is sufficient knowledge to proceed to the next stage of development and research. The Department plans to proceed on a technically conservative basis allowing for ready retrievability of the emplaced waste for some initial period of time.

(a) Disposal of radioactive wastes in mined geologic repositories was stated by the President in his February 12, 1980 message as the interim planning strategy to receive emphasis pending environmental review under NEPA.

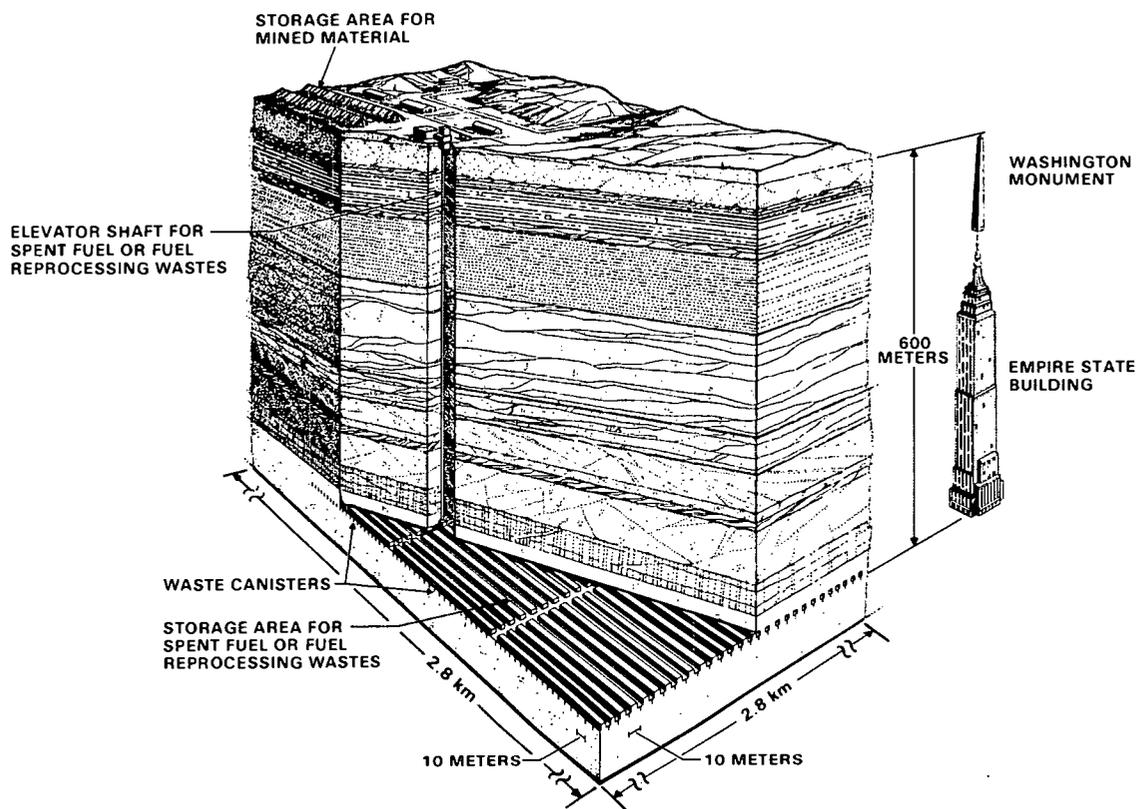


FIGURE 1.3.1 Deep Underground Geologic Waste Repository

1.3.1 Mined Geologic Disposal of Radioactive Wastes

The concept of mined geologic disposal of radioactive wastes is one in which canistered high-level wastes and other wastes in canisters, drums, boxes or other packages, as appropriate to their form, radioactive waste content and radiation intensity, are placed in engineered arrays in conventionally mined rooms in geologic formations far beneath the earth's surface. An artist's rendering of the geologic disposal concept is shown together with more familiar structures for comparison in Figure 1.3.1.

Geologic disposal, as analyzed in this Statement, also employs the concept of multiple barriers. Multiple barriers include both engineered and geologic barriers that improve confidence that radioactive wastes, in biologically significant concentrations, will not return to the biosphere. Engineered barriers include the waste form itself, canisters, fillers, overpacking, sleeves, seals and backfill materials. Each of these components may be designed to reduce the likelihood of release of radioactive material and would be selected based on site- and waste-specific considerations. Geologic barriers include the repository host rock and adjacent and overlying rock formations. While engineered barriers are tailored to a specific containment need, geologic barriers are chosen for their in-situ properties for both waste containment and isolation.

1.3.2 An Example Geologic Repository

For purposes of illustration and for estimating the environmental impacts of development and implementation of waste disposal in geologic repositories, an example repository

was postulated that would have an underground area of about 800 hectares (2000 acres) and would be located about 600 meters (2000 ft) underground. This repository area provides for reasonable waste disposal capacity and is achievable from both construction and operational points of view using conventional room and pillar mining techniques. Actual repositories may be larger or smaller than 800 hectares (ha) depending upon site-specific characteristics.

In this Statement salt, granite, shale and basalt are considered as examples of repository host rock. These rock types represent a range of characteristics of candidate earth materials representative of geologic formations that might be considered but other rock types such as tuff may also be suitable candidates.

Because of restrictions of radioactive waste heat loading on the host rock (to prevent or restrict effects on the rock structure) and other structural considerations, different spacing of waste canisters (containers) would be required and would result in different repository waste capacities for a given rock type and repository area.

The number of 800-ha example repositories required for disposal of spent fuel or reprocessing wastes under the different nuclear power growth assumptions described in Section 1.1 is given in Table 1.3.1. The ranges given reflect the different load capacities (both from a permissible heat load standpoint and because of the different fractions of the 800 ha available for waste emplacement) of repositories in the different host rocks.

TABLE 1.3.1. Number of 800 Hectare Example Repositories Required for Various Nuclear Power Growth Assumptions

Case	Nuclear Power Growth Assumption	Number of Repositories	
		Spent Fuel	Reprocessing Wastes
1	Present Inventory Only Reactors Shut Down in 1980	0.03 to 0.1	(a)
2	Present Capacity and Normal Life	0.2 to 1	(a)
3	250 GWe System by Year 2000 and Normal Life	1 to 4	2 to 5
4	250 GWe System by Year 2000 and Steady State ^(b)	2 to 5	3 to 6
5	500 GWe System by Year 2000 ^(b)	2 to 7	4 to 9

(a) If all reactors are shut down in 1980 or if nuclear power were to be restricted to present capacity there would be no economic incentive for reprocessing.
(b) Required by Year 2040.

As shown in Table 1.3.1 the subterranean area needed for spent fuel or reprocessing wastes from the power-generating scenarios considered in this Statement ranges from approximately 24 ha (60 acres) to about 7,200 ha (18,000 acres or 24 mi²) depending upon the scenario and the choice of repository media. The larger numbers of repositories for reprocessing wastes are required principally because of the large volumes of TRU wastes requiring disposal.

Once licensing approvals are obtained, an approximate 5-year repository construction period is estimated. The operating period may range from 1 to 30 years or more depending on the size of the industry served and on the number of repositories operating concurrently.

1.3.3 Environmental Impacts Associated with Construction and Operation of Example Geologic Repositories

Environmental impacts associated with construction and operation of geologic repositories include radiological impacts, both in the short and long term, land and other resource commitments, and impacts related to ecological, nonradiological, aesthetic, and socioeconomic aspects. In the case of socioeconomic, aesthetic, and ecological impacts and hypothetical failures of repositories in the long term, impacts are summarized for a single 800-ha repository, as might be built in salt, granite, shale or basalt and containing either spent fuel or reprocessing wastes. Radiological impacts of waste management and disposal, resource commitments and dollar costs are summed in Section 1.7 for total system requirements for power growth assumptions given in Table 1.1.1.

1.3.3.1. Radiological Impacts

Radiological impacts that might be associated with repository construction (mining), operation and decommissioning, as well as those that might result from unplanned events either before or after the repository was closed were analyzed in detail. The estimated 70-year whole-body dose to a hypothetical regional population (2 million persons) from radon and radon daughter products as a result of repository mining operations ranges from less than one to 100 man-rem depending on host rock. During the time the repository was receiving wastes (6 to 20 years), normal operations might add about 1 man-rem to this total. During these time periods, the regional population would have received from about 1,000,000 to 4,000,000 man-rem from naturally occurring, undisturbed radionuclides. Thus, construction and operation of a geologic repository under normal conditions do not constitute a significant radiological impact.

Accidents occurring during operation of the repository that might have radiological impacts were also investigated. The accident believed to have the largest potential radiological consequence is the dropping of a waste canister down the repository shaft and rupture of the canister on impact. The 70-year whole-body doses to the regional population from such accidents were determined to total to less than 6000 man-rem for 20 years of waste emplacement in a repository. During the same period the regional population would receive about 4,000,000 man-rem from naturally occurring sources. However, doses to workers in the repository from radioactive material released in the event of a canister drop could be fatal (greater than 7,000 rem in first year following the accident). Engineered precautions similar to those outlined in Section 5.4 are expected to preclude such consequences and to reduce doses to workers to safe levels.

Results of a total system analysis of radiological and other impacts for the various power generating projections are summarized in Section 1.6. For those interested in details of environmental aspects of the complex interactions of predisposal and disposal activities, and power growth assumptions, Chapter 7 should be consulted.

1.3.3.2 Resource Commitments

Various resources would be required in the construction and operation of geologic repositories. Ranges of some of the more important resource commitments, as a function of host rock, are presented in Table 1.3.2. The values given are based on a normalized energy production basis of one GWe-yr (about 9 billion kWh, equivalent to one large reactor operating for one year).

Even at an installed nuclear power capacity of 250 GWe operating over several decades the tabulated material and energy commitments are but a small fraction of that used for the

TABLE 1.3.2 Resource Commitments Associated with Construction and Operation of Geologic Waste Repositories, Normalized to 1 GWe-yr

	<u>Spent Fuel Repositories</u>	<u>Fuel Reprocessing Waste Repositories</u>	<u>Approximate U.S. Annual Production</u>
Propane, m ³	1.6 - 2.0	1.5 - 3.3	1 x 10 ⁶
Diesel Fuel, m ³	1.2 x 10 ² - 1.7 x 10 ²	1.7 x 10 ² - 2.5 x 10 ²	4 x 10 ⁸
Gasoline, m ³	1.2 x 10 ¹ - 1.5 x 10 ¹	1.1 x 10 ¹ - 2.4 x 10 ¹	6 x 10 ⁸
Electricity, kw-hrs	1.0 x 10 ⁶ - 1.1 x 10 ⁶	1.3 x 10 ⁶ - 1.8 x 10 ⁶	2 x 10 ¹²
Manpower, man-yrs	1.6 x 10 ¹ - 1.7 x 10 ¹	1.8 x 10 ⁴ - 3.3 x 10 ¹	4 x 10 ⁶ (a)
Steel, MT	2.5 x 10 ¹ - 6.1 x 10 ¹	6.2 x 10 ¹ - 1.0 x 10 ²	1 x 10 ⁸
Cement, MT	2.2 x 10 ¹ - 2.6 x 10 ¹	2.9 x 10 ¹ - 6.7 x 10 ¹	7 x 10 ⁷
Lumber, m ³	1.7 - 2.1	1.6 - 3.5	3 x 10 ⁹

(a) Construction and mining.

total economy. To give additional perspective to the consumption of energy as fossil fuel and electricity, each was converted to units of energy expended in deep geologic disposal of waste per unit of energy produced by the fuel from which the waste came. In the case of spent fuel 0.04% of the energy produced was consumed in geologic waste disposal and in the case of fuel reprocessing wastes 0.05% of the energy produced was consumed. On this basis it is concluded that the irretrievable commitment of the above materials is warranted.

1.3.3.3 Socioeconomic Impacts

Socioeconomic impacts associated with the construction and operation of repositories are dependent largely on the number of persons who move into the locality in which the facility will be located. Site characteristics that are especially important in influencing the size of the impacts include the availability of a skilled local labor force, secondary employment, proximity to a metropolitan area, and demographic diversity (population size and degree of urbanization) of counties in the commuting region. An additional factor in the generation of impacts is the time pattern of project-associated population change. For

example, a large labor force buildup followed closely by rapidly declining project employment demand could cause serious economic and social disruptions both near the site and within the commuting region.

In this Statement impacts are estimated for three reference sites, identified as Southeast, Midwest, and Southwest. These areas were chosen because siting of facilities in those regions is plausible and because they differ substantially in demographic characteristics, thus providing a reasonable range of socioeconomic impacts.

In general, the reference Southwest site is more likely to sustain significant socioeconomic impacts than are the other two sites, because it has a smaller available unemployed construction labor force, lacks a nearby metropolitan center, and is subject to the generation of greater secondary employment growth than are the other sites. If a repository were to be built in an area where demographic conditions approximated those of the Southwest site, a detailed analysis of site-specific socioeconomic impacts would be needed to help prevent serious disruptions in provision of necessary social services.

Table 1.3.3 presents the manpower requirements for construction and operation of a single waste repository accepting either spent fuel or reprocessing wastes.

TABLE 1.3.3. Manpower Requirements for Construction and Operation of a Single Waste Repository (three peak years)

Repository Medium	Average Annual Employment			
	Spent Fuel Repository		Reprocessing Waste Repository	
	Construction	Operation	Construction	Operation
Salt	1700	870	2000	1300
Granite	4200	1100	3000	1300
Shale	2200	880	2100	1200
Basalt	5000	1100	3800	1500

1.3.3.4 Land Use, Ecological Impacts and Other Impacts

At an 800-ha repository, above ground facilities (including mining spoils piles) would occupy about 200 to 300 ha depending on geologic media. An additional 10 ha would be used for access roads. An 800-ha area above the subterranean repository would be set aside at the surface, and mineral and surface rights would be restricted. This surface land, except that occupied by mining spoils piles, could be returned to its former use when the repository surface facilities are decommissioned after sealing and closure of the repository. Presently an area equal to 3,200 ha, centered over the repository, is considered necessary for exclusion of nearby subsurface activities. Subsurface activities could be restricted as long as institutional control exists. (It is expected that this issue will be more closely examined for site-specific applications. Present plans call for a repository design that does not need to rely on institutional controls after closure.)

The main ecological concern of repository construction and operation is the potential for airborne and waterborne contamination of the environs as a result of the very large mine spoils piles. Land near repositories in salt could be contaminated by windblown salt;

nearby streams could be harmed by runoff contaminated with salt. Removal of the salt to a nonharmful environment, such as through dilute dispersal at sea or stabilization of the salt piles could obviate the problem. Repositories in shale do not appear to pose as serious a problem, although alteration of pyrite, a mineral found in shales, could lead to contamination of streams. The spoils piles from repositories in granite and basalt are not expected to have a significantly adverse affect on the environment.

It is possible that for any rock type the pile of rock left on the surface will have an adverse aesthetic impact. The possibility also exists that these spoils piles of rock (millions of MT), if arranged properly, could become markers identifying the locations of the repositories--although some would maintain that such markers eventually might actually enhance the probability of archaeological exploration.

It is concluded that, in a generic sense, neither land use nor ecological impacts are of such a magnitude as to deter development of geologic repositories or their use for disposal of nuclear radioactive wastes from commercial power generation.

1.3.4 Environmental Impacts in the Long Term

Planned functioning of the geologic repository after closure will result in very little in the way of environmental impacts. So long as institutional controls exist there will probably be some control of land useage above the repository. There will probably be some monitoring performed until future generations decide to discontinue monitoring. Although heat from the waste will ultimately reach the surface over the repository, the estimated temperature rise is expected to be less than 0.5°C in all cases. Small amounts of uplift and subsidence might occur for repositories in salt and shale but probably none for repositories in granite and basalt. During planned functioning of the waste repository after closure there will be no health effects attributable to the repository.

Although waste repositories will be sited, loaded, and sealed with every expectation that long term radiological impacts will be nonexistent, the ways in which a repository might fail, the likelihood of its failure, and the consequences to the human environment of such failure were investigated in detail. At 600 m below the earth's surface, it is extremely improbable that wastes in biologically important concentrations would ever reach the human environment. Nevertheless, several events were postulated that might release repository contents, and estimates were made of the possible consequences of such release, in terms of radiation dose to, and postulated health effects among, the public. In brief, these events were:

- impact of a giant meteorite directly over the repository releasing some of the repository contents to the atmosphere (which is believed to have consequences on the order of other events such as volcanism and nuclear warfare that might breach a repository)
- faulting or other fracturing of the host rock, followed by flooding of the repository with water and either a) contamination of an emergent stream, b) slow ground-

water transport to the biosphere, or c) contamination of a near surface aquifer that had been tapped by a well

- human intrusion by drilling for exploration
- solution mining of salt in the case of a repository in salt.

The doses to the regional population were calculated for each event and then the number of radiation-related health effects was determined by applying a conversion factor of from 100 to 800 health effects (50 to 500 fatal cancers plus 50 to 300 serious genetic disorders) per million man-rem (as developed in Appendix E). The results were then multiplied by the probability (where determinable) that the event would occur, to obtain a measure of expected societal risk.

Societal risk in each case where probabilities could be estimated were very small; for example, in the case of breach by a giant meteorite whose probability was estimated to be 2×10^{-13} /yr and where the largest calculated consequences were 1.4×10^5 health effects, the societal risk amounted to 3×10^{-8} health effects/yr, and in the case of faulting and flooding the societal risk amounted to 3×10^{-11} health effects/yr. For comparison, the expected societal risk from lightning in the population of 2 million, in the reference environment, is about 1 fatality per year. In the worst case of general contamination of water, not more than one radiation-related fatality was projected to result over a 10,000-year period.

Although believed to be highly unlikely because of the extreme depth of the repository, no probability could be assigned to the act of drilling into a repository. If, however, drilling did take place within the surface projection of the repository area and to the depth of the repository, the probability was determined to be 0.005 per 1000 drill holes (based on relative cross-sections and spatial density of canisters in the repository) that a waste canister would be intercepted. If drilling took place about 1000 yrs after disposal and a high-level waste canister were penetrated, the contaminated drilling mud, when brought to the surface, could result in a small increase in risk of adverse health effects occurring among about two dozen people postulated to live in the immediate area, if no cleanup takes place.

Even if drilling into the repository were to occur without canister penetration the drill hole might constitute a conduit for entry of water into the repository. Mechanisms to return the water to the biosphere are more difficult to postulate. Regardless, if this event took place, the consequences are believed to be significantly less than those resulting from faulting and flooding scenarios also discussed in this Statement.

Because of the abundance of salt in this country, and its frequent location at depths much less than 600 m, the chance of solution mining near a repository in bedded salt formations is believed to be remote. However, solution mining in a domed salt formation is

(a) The production rate of the hypothetical salt solution mine was estimated to be sufficient to supply salt for about 40 million people.

believed to be much more likely. Part of the reason for this is that there may be geologic surface features that suggest the presence of domed salt; however these features are absent for deeply bedded salt. Assuming that a repository in salt was breached in the course of solution mining for salt and that salt was mined for one year before it was discovered to be contaminated, doses about one-tenth of those from naturally occurring sources were calculated to result among the 40 million people assumed to be consuming the contaminated salt.^(a) Health effects were also estimated to be about one-tenth of those that might be attributable from natural background.

1.4 ALTERNATIVE ACTION--BALANCED DEVELOPMENT OF ALTERNATIVE DISPOSAL METHODS^(a)

The alternative program strategy calling for balanced development of several alternative methods requires selection of some other disposal alternative(s) in addition to mined geologic repositories. The following disposal methods are analyzed as candidates for consideration in the alternative waste disposal program. and from this analysis, mined geologic, very deep hole, and subseabed disposal are identified as the most likely candidate technologies for balanced development.

1.4.1 Very Deep Hole Waste Disposal Concept

A very deep hole concept has been suggested that involves the placement of nuclear waste in holes in geologic formations as much as 10,000 meters (6 miles) underground. Potential rock types for a repository of this kind include crystalline and sedimentary rocks located in areas of tectonic and seismic stability.

Spent fuel or high-level waste canisters could be disposed of in very deep holes. However, it is not economically feasible to dispose of high-volume wastes (e.g., TRU) in this manner and thus another alternative, such as deep geologic repositories, is also required if spent fuel is reprocessed. There is some question whether or not drilling of holes to the depths suggested and in the sizes required can be achieved.

The principal advantage of the very deep hole concept is that certain (but not all) wastes can be placed farther from the biosphere, in a location where it is believed that circulating ground water is unlikely to communicate with the biosphere.

1.4.2 Rock Melt Waste Disposal Concept

The rock melt concept for radioactive waste disposal calls for the direct placement of liquids or slurries of high-level wastes or dissolved spent fuel, with the possible addition of small quantities of other wastes, into underground cavities. After the water has evaporated, the heat from radioactive decay would melt the surrounding rock. The melted rock has been postulated to form a complex waste form by reaction with the high-level waste. In about 1000 years, the waste-rock mixture would resolidify, trapping the radioactive material in what is believed to be a relatively insoluble matrix deep underground. Since solidification takes about 1000 years the waste is most mobile during the period of greatest fission product hazard.

Not believed to be suitable for rock melt disposal are wastes from reprocessing activities such as hulls, end fittings, and TRU wastes remaining after dissolution. Because of the inability to accommodate these wastes, some other disposal method would have to be used in conjunction with the rock melt disposal concept.

(a) Analyses developed in this Statement under the alternative program evaluate the environmental impacts of deferring implementation of a disposal program until the year 2030. This situation can also be interpreted as demonstrating impacts that would result from a delayed disposal program.

1.4.3 Island-based Geologic Disposal Concept

Island-based disposal involves the emplacement of wastes within deep stable geological formations, much as in the conventionally mined geologic disposal concept and in addition relies on a unique hydrological system associated with island geology. Island-based disposal would accommodate all forms of waste as would conventionally mined geologic disposal; however, additional port facilities and additional transportation steps would be required. Remoteness of the probable candidate islands has been cited as an advantage in terms of isolation.

1.4.4. Subseabed Disposal Concept

It has been suggested that wastes could be isolated from the biosphere by emplacement in sedimentary deposits beneath the bottom of the deep sea (thousands of meters below the surface), which have been deposited over millions of years. The deposits have been shown by laboratory experiments to have high sorptive capacity for many radionuclides that might leach from breached waste packages. The water column is not considered a barrier, however it will inhibit human intrusion and can contribute to dilution by dispersal of radionuclides that might escape the sediments.

One subseabed disposal system incorporates the emplacement of appropriately treated waste or spent reactor fuel in free-fall needle-shaped "penetrometers" that, when dropped through the ocean, would penetrate about 50 to 100 m into the sediments. A ship designed for waste transport and placement would transport waste from a port facility to the disposal site and would be equipped to emplace the waste containers in the sediment.

Subseabed disposal is an attractive alternative disposal technique because technically it appears feasible that, at least for high-level waste and spent fuel, the waste can be placed in areas having relatively high assurance of stability. If at some point in time all of the barriers failed, the great dilution and slow movement should retard the return of radionuclides to the human environment in biologically important concentrations. The research needed to technically permit subseabed disposal to go forward has been projected not to be as costly or time consuming as some other alternatives. On the other hand, like island-based geologic disposal, the subseabed concept has the disadvantage of the need for special port facilities and for additional transportation steps in comparison to mined repositories on the continent.

As noted, subseabed disposal is believed to be technologically feasible; however, international and domestic legal problems to its implementation would require favorable resolution. Whether subseabed disposal can provide isolation of wastes equal to that of deep geologic repositories has not been fully assessed. Because of volume considerations, subseabed disposal does not appear practical for TRU wastes and some other method would be required for their disposal.^(a)

(a) Trenches in the ocean floor have been suggested as a means of disposing of higher volume, but less radioactive wastes.

1.4.5 Ice Sheet Disposal Concept

Disposal in continental ice sheets has been suggested as a means of isolating high-level radioactive waste. Past studies have specifically addressed the emplacement of waste in either Antarctica or Greenland. The alleged advantages of ice sheet disposal, which are disposal in a cold, remote area and in a medium that should isolate the wastes from man for many thousands of years, cannot be proven on the basis of current knowledge.

Proposals for ice sheet disposal of high-level waste and/or spent fuel suggest three emplacement concepts:^(a)

- Passive slow descent--waste is emplaced in a shallow hole and the waste canister melts its own way to the bottom of the ice sheet
- Anchored emplacement--similar to passive slow descent but an anchored cable limits the descent depth and allows retrieval of the canister and prevents movement to the bottom of the sheet.
- Surface storage--storage facility supported above the ice sheet surface with eventual slow melting into the sheet.

Ice sheet disposal, regardless of the emplacement concept, would have the advantages of remoteness, low temperatures, and isolating effects of the ice. On the other hand, transportation and operational costs would be high, ice dynamics are uncertain, and adverse global climatic effects as a result of melting of portions of the ice are a remote possibility. The Antarctic Treaty now precludes waste disposal in the Antarctic ice sheet. The availability of the Greenland ice sheet for waste disposal would depend upon acceptance by Denmark and the local government of the island itself.

A great deal of research appears to be needed before the potential of ice sheet disposal is determined. Even though the apparent bowl-shaped ice cap of Greenland would result in the wastes melting to the bottom of the bowl where they might remain permanently, the consequences of release of radioactive decay heat to the ice are uncertain. Because of weather extremes and environmental conditions on the ice sheets, difficulties are also predicted for transportation of the wastes to the site, waste emplacement and site characterization.

1.4.6 Well Injection Disposal Concepts

Two methods of well injection have been suggested: deep well liquid injection and shale/grout injection.

Deep well liquid injection involves pumping acidic liquid waste to depths of 1000 to 5000 m (3,300 to 16,000 ft) into porous or fractured strata that are suitably isolated from the biosphere by relatively impermeable overlying strata. The waste is expected to remain

(a) Present concepts for waste disposal in ice sheets call for TRU reprocessing waste to be placed in mined geologic waste repositories.

in liquid form and may thus progressively disperse and diffuse throughout the host rock. Unless limits of movement are well defined, this mobility within the porous host media formation would be of concern regarding eventual release to the biosphere.

For the shale/grout injection alternative, the shale is fractured by high-pressure injection and then the waste, mixed with cement and clays, is injected into the fractured shale formations at depths of 300 to 500 m (1000 to 1600 ft) and allowed to solidify in place in a set of thin solid disks. Shale has very low permeability and predictably good sorption properties. The formations selected for injection would be those in which it can be shown that fractures would be created parallel to the bedding planes and in which the wastes would be expected to remain within the host shale bed. This requirement is expected to limit the injection depths to the range stated above.

This alternative is applicable only to reprocessing wastes or to spent fuel that has been processed to liquid or slurry form. Therefore, well injection is not sufficient to dispose of all wastes generated, and a suitable additional technique would be required.

1.4.7 Transmutation Concept

In the reference transmutation concept, spent fuel would be reprocessed to recover uranium and plutonium (or processed to obtain a liquid high-level waste stream in the case where uranium and plutonium are not to be recycled). The remaining high-level waste stream is partitioned into an actinide waste stream and a fission product stream. The fission product stream is concentrated, solidified, and sent to a mined geologic repository for disposal. The waste actinide stream is combined with uranium or uranium and plutonium, fabricated into fuel rods, and reinserted into a reactor. In the reactor, about 5 to 7% of the recycled waste actinides are transmuted to stable or short-lived isotopes, which are separated out during the next recycle step for disposal in the repository. Numerous recycles would result in nearly complete transmutation of the waste actinides; however, additional waste streams are generated with every recycle. Transmutation, however, provides no reduction in the quantities of long-lived fission product radionuclides such as ^{99}Tc and ^{129}I in the fission product stream that is sent to geologic disposal.

1.4.8 Space Disposal Concept

Space disposal has been suggested as a unique option for permanently removing high-level nuclear wastes from the earth's environment. In the reference concept, high-level waste is formed into a ceramic-metal matrix, and packaged in special flight containers for insertion into a solar orbit, where it would be expected to remain for at least one million years. The National Aeronautics and Space Administration (NASA) has studied several space disposal options since the early 1970s. The concept involves the use of a special space shuttle that would carry the waste package to a low-earth orbit where a transfer vehicle would separate from the shuttle and place the waste package and another propulsion stage into an earth escape trajectory. The transfer vehicle would return to the shuttle while the remaining rocket stage inserts the waste into a solar orbit.

Space disposal is of interest because once the waste is placed in orbit its potential for environmental impacts and human health effects is judged to be nonexistent. However, the risk of launch pad accidents and low earth orbit failures have not been determined.

The space disposal option appears feasible for selected long-lived waste fractions of radionuclides such as ^{129}I , or even for the total amount of reprocessed high-level waste that will be produced. Space disposal of unprocessed fuel rods and other high volume wastes does not appear economically feasible or practical because of the large number of flights involved.

1.5 NO-ACTION ALTERNATIVE

The no-action alternative would leave spent fuel or reprocessing wastes at the sites generating the waste or possibly at other surface or near-surface storage facilities for an indefinite time. In this alternative, existing storage is known to be temporary and no consideration has been given to the need for additional temporary storage when facilities in use have exceeded their design lifetime. There seems to be no question but that at some point in time wastes will require disposal and that considerable time and effort will be required to settle upon an adequate means of disposal. It seems clear that development of acceptable means of disposal of wastes is sufficiently complex and of sufficiently broad national importance that coordination of research and development, construction, operation, and regulation at the Federal level is required and that the no-action alternative is unacceptable. Indeed, adoption of a no-action alternative by the Department of Energy could be construed as not permissible under the responsibility mandated to the Department by law. Neither would a no-action alternative be in accord with the President's message of February 12, 1980, when he stated that "...resolving...civilian waste management problems shall not be deferred to future generations."

1.6 PREDISPOSAL SYSTEMS^(a)

After the wastes are generated and before they are disposed of, several predisposal operations are required. The combination of these operations is referred to as a predisposal system. System operations include treatment and packaging to prepare the waste for the specific requirements of a disposal option, interim storage if the treated waste cannot be shipped immediately to a disposal site, shipment to interim storage and/or to a disposal site, and decommissioning of the waste treatment and storage facilities. In considering various alternatives for disposal of wastes, different operations for predisposal treatment required by each alternative must also be compared.

All of the alternatives that utilize a dissolution process would also generate considerable quantities of miscellaneous TRU waste. It is assumed here that these materials are always sent to a mined geologic repository regardless of the disposal option selected for high-level waste.

1.6.1 Predisposal System for the Once-Through Cycle

Following discharge from the reactor, spent fuel is stored for a period of time at reactor storage basins. The fuel is then shipped to a treatment and/or packaging facility if a disposal facility is available. If a disposal facility is not available at the end of the reactor storage period, the fuel is assumed to be shipped to an away-from-reactor (AFR) storage facility and subsequently shipped to available repositories. When a disposal facility is available at the end of the reactor storage period, the fuel is shipped to a treatment and/or packaging facility. If the disposal site is separate from the treatment and/or packaging facility, the fuel is then shipped to the disposal site.

Initial storage and shipment operations are identical for all of the disposal alternatives. The differences imposed on the predisposal systems by the disposal alternatives are in the treatment and/or packaging and final shipment to disposal.

1.6.2 Predisposal System for the Reprocessing Cycle

In the reprocessing cycle, wastes requiring disposal are produced at the fuel reprocessing plant (FRP) and at the mixed-oxide fuel fabrication plant (MOX-FFP). Both high-level waste and TRU waste are produced at the FRP but only TRU wastes are produced at the MOX-FFP. These wastes are assumed to be treated and packaged at the site where they are produced, either the FRP or MOX-FFP. They are then shipped to interim storage if a disposal facility is not available; finally, they are shipped to a disposal facility.

1.6.3 Accident Impact Summary for Predisposal Operations

Table 1.6.1 summarizes the results of the predisposal-system accident analyses. This table shows that transportation is the waste management step with the potential for the

(a) Although this section is very brief, predisposal systems involve many facilities, operations, and processes and for those interested, details are given in Chapter 4.

TABLE 1.6.1. Summary of Radiation Effects from Potential Worst-Case Predisposal System Accidents

	<u>70-Year Dose to Maximum-Exposed Individual, rem</u> <u>Once-Through Cycle</u>	<u>Reprocessing Cycle</u>
Transportation (impact and fire)		
Spent Fuel (4-year-old)	0.6(a)	
HLW		10(b)
TRU Waste		3
Storage	5×10^{-2}	8×10^{-3}
Treatment and Packaging	3×10^{-5}	2×10^{-3}

(a) Shipment of 6-month-old spent fuel, which is unlikely, could result in a maximum individual dose of 130 rem.

(b) The age of HLW at shipment in the scenario used in this Statement would be about 6-1/2 years old.

most serious accidents in either fuel cycle. The estimated exposures in these accidents, however, are not large enough to cause observable clinical effects. Only in the case of an accident involving shipment of 6-month-old fuel was the dose (130 rem) determined to be sufficiently large that the individuals exposed would have a significant increase in probability of developing cancer sometime during their life or of passing on a genetic defect.

1.7 ENVIRONMENTAL IMPACTS OF PROGRAMMATIC ALTERNATIVES FOR THE ONCE-THROUGH AND THE REPROCESSING FUEL CYCLE OPTIONS AND VARIOUS NUCLEAR POWER GROWTH ASSUMPTIONS

To assess and compare the overall impacts of implementing the three programmatic alternatives addressed in this Statement, an analysis was made using a computer simulation of the complete waste management system functioning over the entire post-fission lifetime of a nuclear power system. This analysis considers treatment and disposal of all post-fission high-level wastes (spent fuel or reprocessing HLW), airborne wastes^(a) and transuranic (TRU) wastes including decommissioning wastes. In this analysis all waste management functions are accounted for and all radioactive waste streams are tracked each year from origin through treatment, storage, transport and accumulation in a disposal repository.

Both the once-through cycle and the reprocessing cycle are addressed for the proposed and alternative programmatic actions for the nuclear power scenarios presented in Table 1.1.1. For the no-action alternative, indefinite storage of spent fuel in water basin facilities with no ultimate disposal was assumed and reprocessing is not considered. Only the first three nuclear growth cases are considered for the no-action alternative, because, without disposal, growth of nuclear power beyond year 2000 does not appear credible.

DOE estimates that implementation of the proposed program will result in the establishment of operating geologic repositories within the time range of 1997 to 2006. An exact date of operation, depending on a number of variables, will be determined by the outcome of existing programs. To cover additional contingencies such as an accelerated effort to open a repository or, at the other extreme, additional delays for reasons not yet foreseen, a range of repository startup dates from 1990 to 2010 is considered here. The range of impacts is important in this simulation rather than the specific dates of repository startup.

Implementation of the alternative program would result in extending the time to operation of the first disposal system. This action implies a further period of research and development to bring the development status of the selected disposal alternatives to an approximately equal status with current knowledge regarding geologic disposal. At that time, a preferred technology would be selected and effort would be concentrated on developing this preferred technology with a program similar to the currently planned program for implementing geologic disposal. Thus a substantial time delay is inherent in this alternative. Implementation of this alternative program is simulated by a range of repository startup dates from 2010 to 2030.

In the system analysis, mined geologic repositories are used to simulate the disposal method ultimately selected under the alternative program. (This concept is the only one developed sufficiently to model impacts and costs reasonably well, and any alternative disposal concept that might be selected would only be selected if it did not have significantly greater impacts or costs.) The principal effects of the alternative program implementation are the required interim storage for spent fuel or reprocessing wastes, the additional

(a) Airborne wastes from nuclear power plants are not considered in this Statement because such wastes are considered in the EIS prepared for each nuclear power plant.

transportation to and from this storage and the impacts and costs for these operations. Benefits of the delay inherent in this alternative program include the processing and disposal of older and thus less radioactive and cooler wastes.

Repository startup dates considered in the once-through cycle and reprocessing cycle system simulations are shown in Tables 1.7.1 and 1.7.2, respectively. The range of reprocessing startup dates considered is also shown in Table 1.7.2. To simplify the analysis only a single mid-range repository startup date, year 2000 representing the proposed program and 2020 representing the alternative program, was used for Cases 4 and 5. For the same reason only a single mid-range reprocessing date was used for these cases. However, the same potential range as in the other cases should be inferred for both repositories and reprocessing.

TABLE 1.7.1. Repository Startup Dates Considered in the Once-Through-Cycle System Simulations

<u>Nuclear Power Growth Cases</u>	<u>Proposed Program</u>	<u>Alternative Program</u>	<u>No-Action Alternative</u>
1. Present Inventory Only	1990 to 2010 ^(a)	2010 ^(a) to 2030	None
2. Present Capacity and Normal Life	1990 to 2010 ^(a)	2010 ^(a) to 2030	None
3. 250 GWe System by Year 2000 and Normal Life	1990 to 2010 ^(a)	2010 ^(a) to 2030	None
4. 250 GWe System by Year 2000 and Steady State	2000	2020	--
5. 500 GWe System by Year 2040	2000	2020	--

(a) These cases are the same under both the proposed and alternative programs.

TABLE 1.7.2. Reprocessing and Repository Startup Date Combinations Considered in the Reprocessing-Cycle System Simulations

<u>Nuclear Power Growth Cases</u>	<u>Proposed Program</u>		<u>Alternative Program</u>	
	<u>Reprocessing</u>	<u>Repository</u>	<u>Reprocessing</u>	<u>Repository</u>
1. Present Inventory	NA ^(a)	NA ^(a)	NA	NA
2. Present Capacity and Normal Life	NA	NA	NA	NA
3. 250 GWe System by Year 2000 and Normal Life	1990 1990 2010	1990 ^(b) 2010 ^(b) 2010 ^(b)	1990 2010 1990 2010	2010 ^(b) 2010 ^(b) 2030 2030
4. 250 GWe System by Year 2000 and Steady State	2000	2000	2000	2020
5. 500 GWe System by Year 2040	2000	2000	2000	2020

(a) NA = not applicable. Reprocessing assumed not to be undertaken in these low-growth cases.

(b) These cases are the same under both the proposed and alternative programs.

1.7.1 System Radiological Impacts

Both the regional (reference environment of 2 million persons) and worldwide 70-year whole-body dose accumulations for the proposed program, the alternative program, and the no-action alternative are compared for the once-through cycle in Table 1.7.3. Somewhat higher dose accumulations are indicated for the alternative program than for the proposed program. However, the differences are not large enough to be significant.^(a) The dose accumulation for the no-action alternative is somewhat less than for the other alternatives, but considering the time period involved, the differences are not significant. As would be expected, the dose increases with increasing size of the nuclear systems served.

TABLE 1.7.3. Comparison of 70-Year Whole-Body Dose Accumulations from Normal Operations for the Program Alternatives Using the Once-Through Cycle, man-rem

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal Starting 1990 - 2010)		Alternative Program (Disposal Starting 2010 - 2030)		No-Action Alternative	
		Regional	Worldwide	Regional	Worldwide	Regional	Worldwide
1	Present Inventory Only	36	48	36	48	0.2	4
2	Present Capacity Normal Life	200 to 250	290 to 370	250 to 260	370 to 380	90	160
3	250 GWe System by Year 2000 and Normal Life	940 to 1200	1400 to 1800	1200 to 1300	1800 to 1900	480	800
4	250 GWe System by Year 2000 and Steady State	1400	2100	1800	2600	NA ^(a)	NA
5	500 GWe system by Year 2040	1900	2800	2400	3400	NA	NA
	Dose Accumulation from Natural Radiation Sources	1×10^7	4.5×10^{10}	1×10^7	4.5×10^{10}	1×10^7	4.5×10^{10}

(a) NA = not applicable.

The regional and worldwide 70-year whole-body dose accumulations for the proposed and alternative programs are compared for the reprocessing case in Table 1.7.4. The doses are much larger here than in the once-through cycle. However, the dose from reprocessing is only a small fraction of the naturally occurring dose even in the highest nuclear growth case examined here; i.e., 0.5% of the regional dose and 0.003% of the worldwide dose. The doses from either the proposed program or the alternative program are the same. The regional and worldwide dose is accumulated principally (about 95%) from the waste treatment operations and the same quantities of waste are treated in either alternative--the only difference is that waste production and treatment occur at different times.

(a) Result in less than one additional health effect as will be shown in following tables.

TABLE 1.7.4. Comparison of 70-Year Whole-Body Dose Accumulations from Normal Operations for the Program Alternatives Using the Reprocessing Cycle,^(a) man-rem

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal) Starting 1990 - 2010)		Alternative Program (Disposal Starting 2010 - 2030)		No-Action Alternative	
		Regional	Worldwide	Regional	Worldwide	Regional	Worldwide
1	Present Inventory Only	NA ^(b)	NA	NA	NA	NA	NA
2	Present Capacity and Normal Life	NA	NA	NA	NA	NA	NA
3	250 GWe System by Year 2000 and Normal Life	13,000 to 33,000	580,000 to 970,000	13,000 to 33,000	580,000 to 970,000	NA	NA
4	250 GWe System by Year 200 and Steady State	33,000	1,000,000	33,000	1,000,000	NA	NA
5	500 GWe System by Year 2040	46,000	1,500,000	46,000	1,500,000	NA	NA
	Dose Accumulation from Natural Radiation Sources	1×10^7	4.5×10^{10}	1×10^7	4.5×10^{10}	1×10^7	4.5×10^{10}

(a) Assumed reprocessing startup dates range from 1990 to 2000.

(b) NA = not applicable.

In this Statement, 100 to 800 health effects (50 to 500 total cancers plus 50 to 300 serious genetic disorders) are postulated to occur in the exposed population per million man-rem. Based on this criterion, the program alternatives are compared on the basis of health effects in Table 1.7.5 for the once-through cycle and Table 1.7.6 for the reprocessing cycle.

For the once-through cycle, with the high nuclear growth assumption, the number of health effects range from 0 to 2 on a regional basis and 0 to 3 on a worldwide basis. In the reprocessing case, the number of health effects are larger. For the high nuclear growth assumption, they range from 5 to 37 health effects on a regional basis and from 140 to 1100 on a worldwide basis. However, the health effects calculated to occur over the same period from naturally occurring radioactive sources range from 1000 to 8000 health effects to the regional population and 4×10^6 to 4×10^7 health effects to the worldwide population.

1.7.2 System Resource Commitments

Estimates of major resource commitments for construction and operation of the entire waste management system were developed for each of the nuclear growth assumptions and each repository and reprocessing startup date. The resources considered include steel, cement, diesel fuel, gasoline, propane, electricity and manpower.

TABLE 1.7.5. Comparison of Health Effects for the Program Alternatives Using the Once-Through Cycle

Case	Nuclear Power Growth Assumption	Number of Effects				No-Action Alternative	
		Proposed Program (Geologic Disposal Starting 1990 - 2010)		Alternative Program (Disposal Starting 2010 - 2030)		Regional	Worldwide
		Regional	Worldwide	Regional	Worldwide		
1	Present Inventory Only	0	0	0	0	0	0
2	Present Capacity and Normal Life	0	0	0	0	0	0
3	250 GWe System by Year 2000 and Normal Life	0 to 1	0 to 2	0 to 1	0 to 2	0	0 to 1
4	250 GWe System by Year 2000 and Steady State	0 to 1	0 to 2	0 to 1	0 to 2	NA(a)	NA
5	500 GWe System by Year 2040	0 to 2	0 to 2	0 to 2	0 to 3	NA	NA

(a) NA = not applicable.

TABLE 1.7.6 Comparison of Health Effects for the Program Alternatives Using the Reprocessing Cycle

Case	Nuclear Power Growth Assumption	Number of Effects				No-Action Alternative	
		Proposed Program (Geologic Disposal Starting 1990 - 2010)		Alternative Program (Disposal Starting 2010 - 2030)		Regional	Worldwide
		Regional	Worldwide	Regional	Worldwide		
1	Present Inventory Only	NA(a)	NA	NA	NA	NA	NA
2	Present Capacity and Normal Life	NA	NA	NA	NA	NA	NA
3	250 GWe System by Year 2000 and Normal Life	1 to 26	6 to 750	1 to 26	6 to 750	NA	NA
4	500 GWe System by Year 2040	3 to 27	100 to 800	3 to 27	100 to 800	NA	NA
5	500 GWe System by Year 2040	5 to 37	140 to 1100	5 to 37	140 to 1100	NA	NA

(a) NA = not applicable.

For the proposed program, resource requirements for reprocessing are somewhat higher than for the once-through cycle in the case of steel, cement, electricity, and manpower; are about the same to somewhat higher for diesel fuel and gasoline; and are substantially higher for propane. The higher propane requirement results from incineration of combustible waste. Gasoline and diesel fuel are used primarily in transportation. These fuel requirements are based on present practice and can be expected to change as fuel use patterns change generally. The propane requirements for the reprocessing cycle represent about 0.5% of the total U.S. consumption for the period to year 2050 assuming current consumption rates hold constant. The largest diesel fuel use amounts to about 1% of total U.S. consumption over the period.^(a) Electricity consumption amounts to 0.02 to 0.05% to the total energy generated by the nuclear power system in this case.

The resource commitments for the program alternatives using the once-through cycle increase as the size of the nuclear system served increases. With the exception of the present inventory case which changes only slightly, requirements for the alternative program compared to the proposed program tend to range up to 2 to 3 times higher for steel, cement, gasoline, propane, and manpower and modestly higher for diesel fuel and electricity. Requirements for the no-action alternative are zero in the present inventory case and are about the same as the alternative program for steel, cement, gasoline, propane, and manpower but diesel and electricity consumption are much lower.

Resource commitments for the program alternatives in the reprocessing cycle tend to be about the same to somewhat higher than for the proposed program requirements.

1.7.3 Systems Costs^(b)

Both total cost and levelized^(c) unit costs (per kWh) were developed. These costs include all waste treatment, storage, transport and disposal costs for wastes resulting from nuclear power generation through the year 2040. The costs also include DOE's research and development and repository site qualification costs which are assumed to be recovered through fees charged to the utilities for storage and disposal. The cost ranges consider four different disposal media.

In terms of total costs, the costs increase with increasing size of the nuclear system but are disproportionately high for the very low-growth cases. The estimated costs range from \$5 to \$12 billion for the present inventory case (Case 1), to \$80 to \$150 billion for the system that reaches 500 GWe installed capacity in the year 2040 (Case 5). Of these totals, the estimated R&D and multiple-site qualification costs range from \$2.9 to \$3.6 billion at the low end of the proposed program to \$9 to \$10 billion at the high end of the

(a) While a commitment of 1% of current U.S. consumption may appear small, some commenters on the draft Statement viewed such a quantity as excessively large in terms of commitment for a single industrial use. It should be noted that resource needs have been approximated for this final Statement. It is believed that optimizing, for instance in terms of shipping distances, could result in reduction of quantities of resource required.

(b) All costs are cited in terms of 1978 dollars.

(c) Levelized Unit Cost = $\frac{\text{Annualized Capital and Operating Costs}}{\text{Annualized Units Produced}}$.

alternative program. The range of costs for the alternative program is higher than the proposed program for the once-through cycle but about the same for the reprocessing cycle. Costs for the no-action alternative are about the same as the low end of the range for the proposed program.

The costs can be better placed in perspective when shown as unit costs per kWh of generated electrical energy. The levelized unit costs are sensitive to the discount rate used (cost of money). Because waste management costs are incurred after the generation of the electricity, increasing the discount rate has the effect of reducing the unit cost. A range of discount rates from 0 to 10% is considered in this Statement and a 7% rate was selected for illustration in this summary. Since the unit cost for the once-through cycle and the reprocessing cycle are similar, the unit costs for the program alternatives are compared in Table 1.7.7 without distinguishing the cost range for each fuel cycle. Costs are somewhat higher when a 0% discount rate is used and slightly lower with a 10% discount rate. On this basis there is little difference between the proposed program and alternative program costs. Cost of electricity in 1978 averaged 3.5 ¢/kWh over all types of services throughout the U.S. On that basis the additional cost for waste management and disposal would add about 2 to 6% to the consumer's cost of electricity and no more than 3% if nuclear power growth to at least 250 GWe is realized.

TABLE 1.7.7. Comparison of Levelized Waste-Management Unit Costs for the Program Alternatives at a 7% Discount Rate, ¢/kWh

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)	No-Action Alternative
1	Present Inventory Only	0.2	0.2	0.08
2	Present Capacity and Normal Life	0.1	0.1	0.06
3	250 GWe system by Year 2000 and Normal Life	0.06 to 0.09	0.07	0.05
4	250 GWe System by Year 2000 and Steady State	0.07 to 0.08	0.07	NA ^(a)
5	500 GWe System by Year 2040	0.06 to 0.08	0.07	NA

(a) NA = not applicable.

1.8 CONCLUSIONS

Based on the environmental impacts evaluated in this Statement, it is concluded that a decision to proceed with the proposed action, that is, development of a programmatic strategy favoring the disposal of commercially generated radioactive wastes in deep geologic repositories, is warranted. This conclusion applies whether the wastes are generated in the once-through or in the reprocessing fuel cycle option.

This conclusion is based on the information contained within this document (and appropriate references) which indicate that the environmental impacts of the program alternatives are similar. The consequences of delaying implementation of a specific disposal technology should not result in any appreciable change in the near-term environmental effects. The decision to emphasize mined geologic repositories as the primary disposal technology is similarly based on an evaluation of the long term effects which indicates that mined geologic disposal and those technologies which justify further consideration would have relatively equal environmental impact. It is recognized that although the level of knowledge of the alternative technologies is not comparable, sufficient evidence exists to support that there is little likelihood that these technologies would be superior, from an environmental perspective, to the geologic alternative.

The no-action alternative is undesirable because of the temporary nature of present storage of wastes, the need to construct additional facilities for extended storage as present facilities reach their design lifetime, and because the no-action alternative is contrary to the presidential proclamation and could be construed as contrary to the mandate given DOE by law. Analysis of the no-action alternative in this Statement has not considered possible failures that could occur if present facilities designed for temporary use were to be used indefinitely. It is possible that no-action could result in unacceptable safety and environmental consequences.

More specifically, regarding the three program alternatives considered in the Statement, the following conclusions can be drawn:

- Radiation dose accumulations increase as the size of the nuclear system increases. Neither the dose accumulation nor the health effects are significantly different for the program alternatives in either the once-through or reprocessing cycles. The dose accumulation with reprocessing is much larger (principally because of doses from radioactive material in dissolver off gas that is released to the environment) (a) than with the once-through cycle. For comparison, this amounts to 0.5% of the regional and 0.003% of the worldwide dose from natural causes over the same period in the highest nuclear growth case examined here.

(a) Estimated dissolver off gas releases are within the EPA Standard for ^{85}Kr and ^{129}I which becomes effective in 1983 (40CFR190.10).

- Resource commitments also increase with increasing size of the nuclear system. With the once-through cycle, resource requirements for the alternative program range up to 2 to 3 times higher than for the proposed program. With the reprocessing cycle, resource requirements for the alternative program are about the same to slightly higher than for the proposed program. For all cases, resource requirements are a small fraction of current U.S. production rates.
- Waste management costs increase as the size of the nuclear system increases, the waste management cost range is significantly higher for the alternative program than for the proposed program. With the reprocessing cycle, the cost ranges are about the same for both alternatives. The no-action alternative costs fall in the low end of the cost range for the proposed program with the once-through cycle. When costs are compared on the basis of levelized unit costs at a 7% discount rate, differences between the alternative and proposed programs and differences between reprocessing and the once-through cycle are slight.
- Societal risk from several events with low probability and high consequence in the long term following geologic repository closure was determined to be small in comparison to other societal risks even if large errors in judgement of the probability of occurrence were made. This conclusion appears valid even if no credit is taken for effects of multiple engineered and geologic barriers that will be employed to further assure containment and isolation.

With respect to the alternative waste disposal technologies considered in this Statement, the following conclusions can be drawn:

- A mined geologic repository is the preferred alternative based on evaluation of radiological effects during the operational period, non-radiological effects on the human environment, status of development, conformance with existing National and international law, independence from future development of the nuclear industry and potential for corrective or mitigating actions. The potential for and consequences of unplanned events in the long term require further investigation. The only category in which an alternative technology might offer an advantage would be the radiological effects during the post-operational period for which space disposal appeared more preferable. However, this long term advantage would be more than offset by near term disadvantages.
- Subseabed disposal appears promising enough to warrant further detailed examination. The potential for and consequences of unplanned events in the long term also require further investigation for this option. Studies of the anticipated environmental

(a) This disposal technology would not be capable of accommodating the full range of waste types. An alternative technology, i.e., geologic disposal, would be required for large quantities of solid waste. Thus, this alternative should be viewed as complementary to geologic disposal.

effects associated with special port facilities and transportation links will be made. The practicality of pursuing this concept, recognizing current National and international laws and agreements will be further analyzed.

- Very deep hole disposal warrants some additional study as a possible backup for HLW disposal only. Further development should emphasize the ability for corrective or mitigating actions available.^(a)
- Space disposal may be profitably studied for its application to special disposal concerns, e.g., more remote isolation of long lived and environmentally mobile radionuclides such as ^{99}Tc and ^{129}I .^(a) However, the overall impact on the total waste management system will need to be carefully evaluated to determine if such separation would provide overall benefit.
- Other technologies studied (island, mined repository, transmutation, rock melt, ice sheet and well injection) either have no clear advantage over geologic disposal, or provide no additional complementary function and, in some cases, are clearly less desirable.

It can be argued that a delay in the program strategy, which would allow for a longer period of R&D, could conceivably reduce the probability of failure of the chosen disposal system by producing more knowledge and a greater diversity of choice in selecting a disposal method. DOE concludes that the likelihood of this occurring is small. In addition, the DOE program allows for a continuing broad based R&D effort, the investigation of a broad range of alternative media, and technical conservatism in program implementation.

Because this Statement is not site-specific it will be necessary to make other environmental analyses addressing the possibility of adverse impacts associated with specific sites and facilities at such time as the program reaches such decision points.

Recovery of the full costs of research and development and implementation of waste management and disposal for all modes of operation considered in this EIS, with the assumption of continued nuclear power growth to 250 GWe, resulted in a 2 to 3% increase in estimated average cost of electrical energy to the consumer. (Complete cessation of nuclear power generation at the end of 1980 would result in a significantly higher cost of waste management per unit of power produced.)

In summary, there appear to be no environmental issues that would reasonably preclude pursuit of a program strategy favoring disposal of commercially generated radioactive wastes in deep geologic repositories (regardless of nuclear power growth assumptions). Thus the proposed action of conducting R&D leading to disposal of radioactive wastes in deep geologic repositories is believed to be fully supported.

(a) This disposal technology would not be capable of accommodating the full range of waste types. An alternative technology, i.e., geologic disposal, would be required for large quantities of solid waste. Thus, this alternative could be viewed as complementary to geologic disposal.

REFERENCES FOR CHAPTER 1

Code of Federal Regulations. Title 40, Part 190.

U.S. Department of Energy. 1980. Statement of Position of the United States Department of Energy in the Matter of Proposed Rulemaking on the Storage and Disposal of Nuclear Waste. DOE/NE-0007, Washington, D.C.

CHAPTER 2

INTRODUCTION

The United States Department of Energy (DOE) has the responsibility to develop technologies for management and disposal of certain classes of commercially generated radioactive wastes (namely high-level and transuranic). To provide input to the decision on a planning strategy for disposal of these radioactive wastes, this Statement presents an analysis of environmental impacts that could occur if various technologies for management and disposal of such wastes were to be developed and implemented.

In this Statement, which often has been referred to as a generic environmental impact statement (GEIS), the various options for permanent waste isolation are examined in a generic or general sense rather than in a site-specific sense. Various concepts are examined for the environmental impacts that their implementation might cause at any non-specific or generic locations. Upon selection of specific locations for waste disposal using the proposed approach, future site-specific environmental analyses will be prepared.

Section 2.1 describes the relationship of this environmental impact statement to other waste management decisions and associated environmental impact statements. This section also outlines the relationship of the President's recent message on disposal of radioactive wastes to the forthcoming National Plan for Nuclear Waste Management.

Section 2.2 describes the structure and content of this Statement. This section also describes the relationship of this Statement's format to those decisions that are to be made (for which this EIS will serve as the environmental input).

Section 2.3 discusses future decisions related to the disposal of commercial radioactive waste.

2.1 RELATIONSHIP TO OTHER WASTE MANAGEMENT DECISIONS

This Statement, Management of Commercially Generated Radioactive Waste, analyzes impacts of high-level and transuranic waste management following removal of spent light water reactor fuel^(a) from nuclear power plants (reactors). The responsibility for developing technology for disposal of radioactive wastes has been assigned to the DOE by the U.S. Congress. The primary emphasis of this Statement is on the safe, permanent isolation of radioactive wastes. Also discussed are interim waste storage, treatment, transportation and facility decommissioning as they relate to a decision on the proposed method of waste disposal.

The basic waste management steps in the commercial LWR nuclear fuel cycle are shown in Figure 2.1.1. The heavy solid lines show waste streams covered in this Statement. Airborne

(a) All but one of the large commercial power reactors operating in the U.S. today are of the light water reactor (LWR) type.

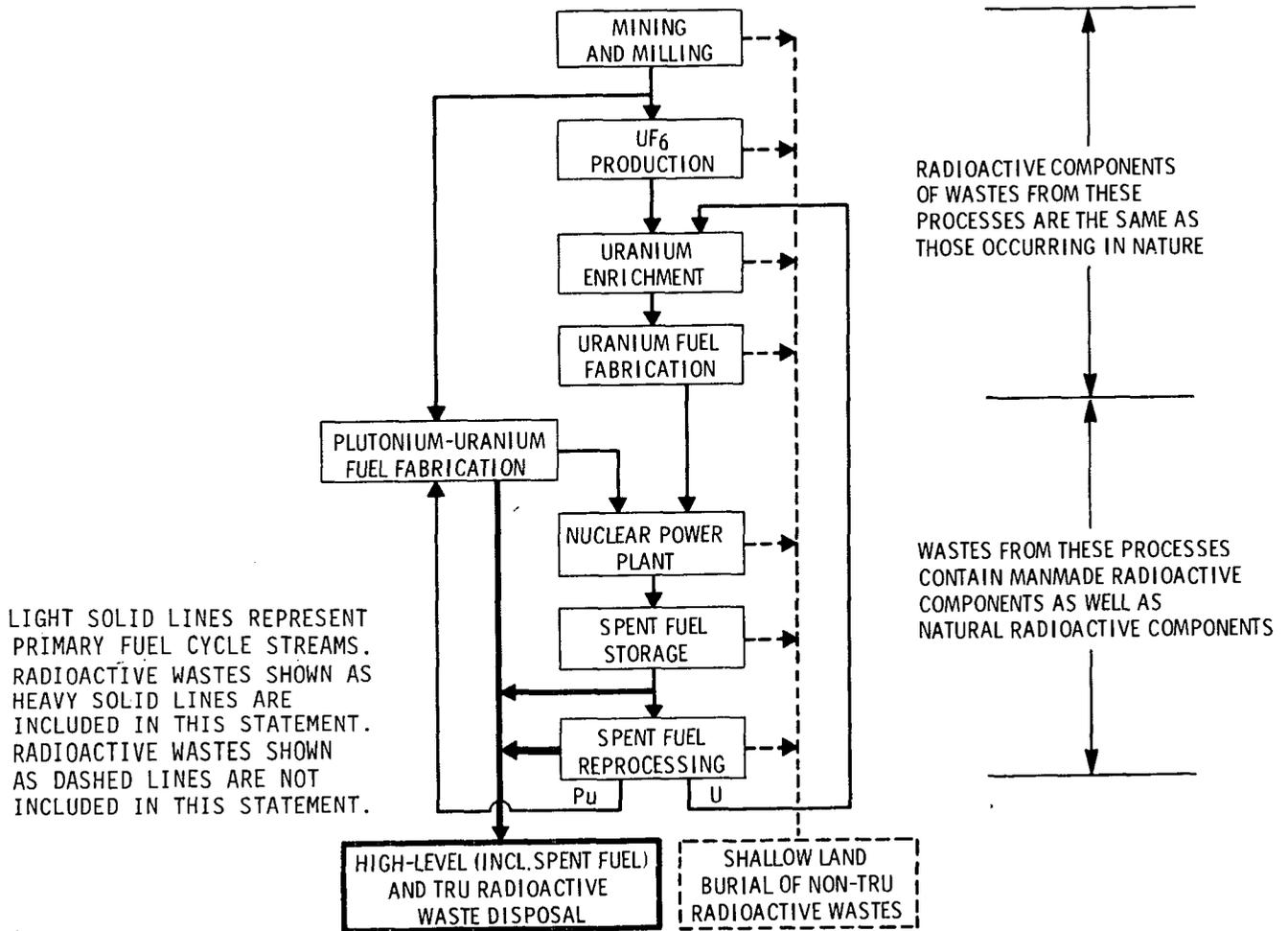


FIGURE 2.1.1. Processes and Waste Streams in the Commercial Fuel Cycle

wastes from spent fuel storage, reprocessing and plutonium-uranium fuel fabrication are also covered. In addition to these wastes, a number of other radioactive wastes must be properly managed and disposed. This section describes the status of program and environmental statements covering these other wastes and also the status of statements covering broad areas (e.g., spent fuel storage and transportation) that are partially included in the overall system addressed in this Statement.

2.1.1 Mining and Milling

Mining and milling operations are currently regulated by either the Nuclear Regulatory Commission (NRC) or by Agreement States (states which have entered into an agreement with NRC pursuant to Section 274 of the Atomic Energy Act of 1954 as amended (42 U.S.C. 2021) under which the state government assumes regulatory authority and responsibility). Environmental impacts are considered programmatically in Uranium Milling, NUREG-0511 (NRC 1979a). Individual EISs have been prepared for each operation licensed. An example is Final Environmental Statement Related to the Plateau Resources Limited Shootering Canyon Uranium Project, NUREG-0583 (NRC 1979b).

2.1.2 Uranium Enrichment

To date, two impact statements have been prepared relative to uranium enrichment:

Final Environmental Statement, Expansion of U.S. Uranium Enrichment Capacity, ERDA-1543 (ERDA 1976)

Final Environmental Impact Statement, Portsmouth Gaseous Diffusion Plant Site, Piketon, Ohio, ERDA-1555 (ERDA 1977a).

2.1.3 Uranium Fuel Fabrication

No generic statement has been prepared for uranium fuel fabrication. This operation is covered by individual statements for specific facilities. Examples of such impact statements are:

Environmental Impact Appraisal, Westinghouse Nuclear Fuel Columbia Site Commercial Nuclear Fuel Fabrication Plant, Columbia, South Carolina, April 1977.

Environmental Impact Appraisal of Nuclear Fuel Services Erwin Plant, Erwin, Tennessee, January 1978.

2.1.4 Low-Level Waste

At present, low-level wastes are regulated by the NRC or by Agreement States. In the event legislation is passed giving DOE any responsibilities related to disposal of low-level wastes from commercial activities, a programmatic environmental statement would be prepared. Environmental impacts of low-level waste activities are described in various NRC documents such as Final Generic Environmental Statement on the Use of Recycled Plutonium in Mixed Oxide Fuel in Light-Water Cooled Reactors, NUREG-002 (NRC 1976).

2.1.5 Spent Fuel Storage

In October 1977, DOE announced a Spent Fuel Storage Policy for nuclear power reactors. Under this policy, U.S. utilities would be given the opportunity to deliver spent power reactor fuel to the U.S. Government in exchange for payment of a fee. The U.S. Government would also be prepared to accept a limited amount of spent fuel from foreign sources when such action would contribute to meeting U.S. nonproliferation goals. A bill was submitted to Congress to authorize action required to implement the Spent Fuel Storage Policy. This bill, known as the "Spent Nuclear Fuel Act of 1979," would authorize the Secretary of Energy to acquire or construct one or more away-from-reactor (AFR) storage facilities. The Secretary would be authorized to accept title to and provide interim storage and ultimate disposal for domestic spent fuel and limited amounts of foreign spent fuel. A final programmatic EIS, Final Environmental Impact Statement, U.S. Spent Fuel Policy, DOE/EIS-0015 (DOE 1980a) has been issued which addresses the environmental impacts of various options regarding the interim storage of domestic fuel, the receipt of some foreign fuel, and the fee methodology for determining the charge for spent fuel storage.

With regard to receipt and storage of foreign spent fuel, the impacts described in the present Statement cover a range of future domestic power production which is sufficiently broad that it would encompass any possible impact due to quantities of spent fuel which might be shipped from other countries to the U.S. Foreign spent fuel which could be returned to the United States for storage or possible disposal would be predominately the LWR type.

Because a decision has been made to implement the Spent Fuel Storage Policy if authorized by Congress, an AFR spent fuel storage facility EIS will be prepared to provide the environmental input into the selection of facilities to meet the demand for spent fuel storage.^(a) The environmental effects associated with the acquisition, construction and/or operation of the facilities and the transportation effects associated with the available options would be evaluated in this environmental documentation.

2.1.6 Transportation

The NRC and the Department of Transportation (DOT) regulate the transportation of radioactive waste. Transportation and packaging criteria and standards are outlined in the Code of Federal Regulations (10 CFR 71 and 49 CFR 170-189). The environmental impacts of transportation activities are addressed in Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes, NUREG-0170 (NRC 1977).

The present Statement specifically examines the transportation of post-fission wastes (spent fuel, high-level waste and TRU waste) from commercial LWR fuel cycle facilities to both interim storage locations and final isolation sites.

2.1.7 Alternative Reactor Types

The present Statement discusses and compares the characteristics of the wastes generated in the management of thorium fuels from the Light Water Breeder (Conversion) Reactor and High-Temperature Gas Cooled Reactor fuel cycle with those obtained from the LWR fuel cycle. No decisions to construct such reactors would be made before consideration is given to the disposal of waste from these reactors. However, the impact of wastes which would be generated by a future Liquid Metal Fast Breeder Reactor (LMFBR) fuel cycle is not analyzed here. They were addressed in Final Environmental Statement, Liquid Metal Fast Breeder Reactor Program, ERDA-1535 (1975a).

2.1.8 Wastes From National Defense Activities

High-level waste from national defense activities is currently being stored on DOE reservations in Idaho, South Carolina, and Washington. EISs that consider the short term storage of these wastes at these sites have been prepared (ERDA 1975b, 1977b, and 1977c, respectively).

(a) The Notice of Intent regarding preparation of the spent fuel storage facility EIS was issued in the Federal Register on August 15, 1980 (45FR54399).

Since waste forms and conditions are different at the three sites, programmatic statements covering development programs for final waste treatment and final disposal are being prepared for each site.

Transuranic wastes resulting from national defense activities are also stored at the sites listed above and at Los Alamos, New Mexico; the Nevada Test Site; and the Oak Ridge National Laboratory in Tennessee. Statements covering waste treatment and final disposal of material now stored at these sites will also be prepared.

This Statement does not directly address management and disposal of radioactive wastes related to national defense programs. However, in a generic sense, systems that can adequately dispose of commercial radioactive wastes have the capability to adequately dispose of wastes resulting from defense programs.

2.1.9 National Plan for Nuclear Waste Management

The President, in his nuclear waste policy statement of February 12, 1980, stated that the safe disposal of radioactive waste, generated from both national defense and commercial activities, is a national responsibility. In fulfillment of his responsibility, the President has directed the Department of Energy, in its role as lead agency for the management and disposal of radioactive wastes, to prepare a comprehensive National Plan for Radioactive Waste Management. This National Plan is being prepared in cooperation with other involved Federal agencies, primarily the Departments of Interior and Transportation, the Environmental Protection Agency, and the Nuclear Regulatory Commission. The State Planning Council, which was established by the President, will also be involved in the development of the National Plan.^(a) This Plan will provide a road map for all parties and give the public an opportunity to review DOE's entire program. The Plan will be comprehensive in scope and include relevant activities of the Federal agencies, states, and local governments. The Plan will cover all types and sources of radioactive waste and present the strategy and sequence of events to manage effectively and dispose of radioactive wastes and associated regulatory activities.

Methods of communication between and among Federal agencies, states and local governments, and the general public will be presented to show current and proposed interactions and the nature and degree of public participation in the planning and decisionmaking process, including the preparation of the National Plan. The National Plan will be updated every 2 years in recognition of and response to results of R&D programs, actual operations, and guidance from institutions such as Federal agencies, state governments, the State Planning Council and others that might be affected by programs and proposed actions.

A draft of the comprehensive National Plan will be distributed by the Secretary of Energy in the fall of 1980, for congressional and general public review and comment. After reviewing public comments and revising the National Plan, a final version of the National

(a) The Council will provide advice and recommendations to the President and the Secretary of Energy on nuclear waste management issues.

Plan, including a summary of the public comments, will be issued in 1981. The National Plan will be used by the Congress, Federal agencies, and the general public to understand the scope, direction, and interrelationship of activities and the progress being made to implement the President's policy.

2.2 STRUCTURE AND CONTENT OF STATEMENT

This Statement describes the character and quantities of the wastes to be managed from various nuclear power generation scenarios and identifies the environmental impacts (i.e., radiological effects, non-radiological effects, resource requirements, socioeconomic impacts, costs, institutional issues) associated with the management of these wastes. The power generation scenarios considered and the scope of the analysis are detailed in Section 3.2. As DOE has the responsibility for selecting a programmatic strategy for the management of commercial radioactive wastes, this Statement presents an analysis of alternative waste management programs for meeting this requirement. The three programmatic strategies presented in the Statement are:

- Proposed Action. The research and development program for waste management will emphasize use of mined repositories in geologic formations in the continental U.S. capable of accepting radioactive wastes from either the once-through or reprocessing cycles (while continuing to examine subseabed and very deep hole disposal as potential backup technologies). This action will be carried forward to identify specific locations for the construction of mined repositories. The proposed action does not preclude further study of other disposal techniques. For example, the selective use of space disposal for specific isotopes might be considered.
- Alternative Action. The research and development program would emphasize the parallel development of several disposal technologies. This action implies an R&D program to bring the knowledge regarding two or three disposal concepts and their development status to an approximately equal level. Based upon the Department of Energy's current evaluation, the likely candidate technologies for this parallel development strategy would be:
 - 1) geologic disposal using conventional mining techniques
 - 2) placement in sediment beneath the deep ocean (subseabed)
 - 3) disposal in very deep holes.

At some later point, a preferred technology would be selected for construction of facilities for radioactive waste disposal.

- No Action Alternative. This alternative would eliminate or significantly reduce the Department of Energy's research and development programs for radioactive waste disposal. Under this alternative, existing spent fuel would be left indefinitely where it is currently stored and any additional spent fuel discharged from future operation of commercial nuclear power plants would likewise be stored indefinitely in water basin facilities either at the reactors or at independent sites.

Beyond the selection of a program strategy, DOE must determine the pace and manner in which to pursue the selected program. To this end, this Statement examines 1) a range of dates for the availability of a mined geologic repository and 2) a variety of candidate repository media (salt, basalt, granite, shale).

The main body of the text (Volume 1) is divided into eight chapters. Chapter 3 presents the program alternatives under consideration and outlines the technological and environmental bases for the analysis. Discussions of natural background radiation and the concept of risk are included to give the reader additional perspectives from which to view the material in the Statement. Non-technical concerns relevant to waste management are also identified for the purpose of airing such issues, which will have to be addressed in any ongoing plan.

Chapter 4 describes the wastes and analyzes the various activities required prior to final disposal on a unit basis (e.g., per GWe-yr, per Kg HM, per facility). The processes of waste treatment, storage, transportation and facility decommissioning are addressed and their impacts are presented. Chemical resynthesis and partitioning, items included in the draft in the presentation of disposal techniques, now appear in the discussion of waste treatment alternatives. A discussion of the relationship between predisposal activities and the individual disposal technologies is also included in Chapter 4.

Chapters 5 and 6 examine the mined geologic disposal concept and alternative disposal technologies, respectively. For consistency of presentation, discussion of each disposal concept addresses the same topic areas:

- Concept and System Description
- Status of Technical Development and R&D Needs
- Disposal Facility Description
- Environmental Impacts of Construction and Operation
- Environmental Impacts Over the Long Term
- Cost Analysis
- Safeguard Requirements.

The depth of the presentation, however, is not identical for the various disposal alternatives for two reasons. First, the extent to which a disposal concept can be examined is a function of the degree to which the concept has been researched, developed, and reported in previous studies. Accordingly, mined geologic disposal is more fully described than the other disposal modes. Secondly, an assessment of the impacts from implementing a disposal alternative is predicated on having data that can be substantiated. The existing data base for mined geologic disposal is significantly more extensive than for the other concepts; hence, a more detailed analysis of impacts is possible.

At the end of Chapter 6, a comparison is made of the nine disposal technologies presented in Chapters 5 and 6 on the basis of several environmental and policy-related criteria.

Chapter 7 outlines the trade-offs between the program alternatives (identified in Chapter 3), with emphasis on the entire waste management system. The points of comparison of the alternative actions deal with nuclear power growth assumptions, fuel cycles, waste volumes, and environmental impacts based on the material in Chapters 4, 5, and 6.

Chapter 8 is a glossary of key environmental, geologic, and waste technology-related terms and acronyms.

Volume 2 is a compilation of appendix material. Volume 3 is a presentation of written public and agency comments and Hearing Board recommendations on the draft Statement and responses to these comments and recommendations.

During the reviews of the draft Statement, some commenters urged that the option of shutting down all nuclear power plants be considered in the final Statement. Although such an action is beyond the authority of the DOE and can be considered only by the NRC or by the U.S. Congress, this Statement does present an analysis of managing only present inventories of spent fuel. While the availability of adequate waste management methods should be considered by these institutions in contemplating such an action, many other far-broader issues, such as national energy and economic requirements and the overall safety and environmental impacts of other energy systems, would also need to be considered. Due to the extent of DOE's authority, the scope of this environmental impact statement is limited to consideration of the impacts of successfully implemented programs for research and development leading to permanent disposal of present and future high-level and TRU radioactive wastes.

2.3 OTHER DECISIONS CONCERNING DISPOSAL OF COMMERCIAL WASTES^(a)

The decisions that the DOE now faces and for which the analysis in this Statement will provide environmental input will not automatically lead to the placement of radioactive wastes in any specific location. As the program of research and development and examination of specific candidate locations proceeds, further decisions will be required relative to potential environmental impacts.

The National Environmental Policy Act of 1969 (NEPA 1969), as implemented by the regulations of the Council on Environmental Quality (CEQ 1978) and the DOE guidelines (DOE 1980b), requires that environmental consequences be considered in Department planning and decisionmaking. In adopting a strategy for disposal of high-level radioactive wastes, the DOE will undertake actions having potential environmental consequences. The potential environmental effects of these actions and their significance vary. Actions range from the decision adopting the overall strategy for waste disposal (involving a major resource commitment which ultimately may have a spectrum of potential environmental effects specific to that strategy) to the selection of specific sites and facilities for waste disposal purposes. Other actions include the conduct of research (data gathering and analysis) which may have little environmental effect but which may have important technological, cost, and time implications on long-term waste disposal.

Using the CEQ regulations and the DOE guidelines, a NEPA implementation plan, which is integrated with overall DOE planning and decisionmaking, has been developed for the deep mined geologic disposal strategy. Figure 2.3.1 graphically demonstrates the various steps associated with integration of the NEPA plan and the overall decisionmaking process.

The DOE's NEPA implementation plan is based on the "tiered" approach, which is designed to eliminate repetitive discussions of the same issues and to focus on the actual issues ripe for decision at each level of environmental review. This approach allows coverage of general matters in broad environmental impact statements (EISs) with subsequent narrower EISs or environmental assessments (EAs) incorporating by reference the general discussions and concentrating solely on the issues specific to the subsequent decision.

The NEPA implementation plan identifies the major decision points in the program to assure that appropriate environmental documentation is completed prior to each such decision and prior to the conduct of activities that may cause an adverse environmental impact or limit the choice of reasonable alternatives. The first major decision process is selection of a program strategy for disposal of nuclear waste. This Statement serves as the NEPA input for this first decision.

(a) Much of the material in this section was taken from the recent DOE Statement of Position in the NRC rulemaking proceedings on nuclear waste storage and disposal (DOE 1980c). The Statement of Position described in DOE's proposed research and development program and was prepared pursuant to the initiation of the rulemaking proceedings. The present Statement, upon issuance as a final impact statement, will become part of the record of the rulemaking proceedings.

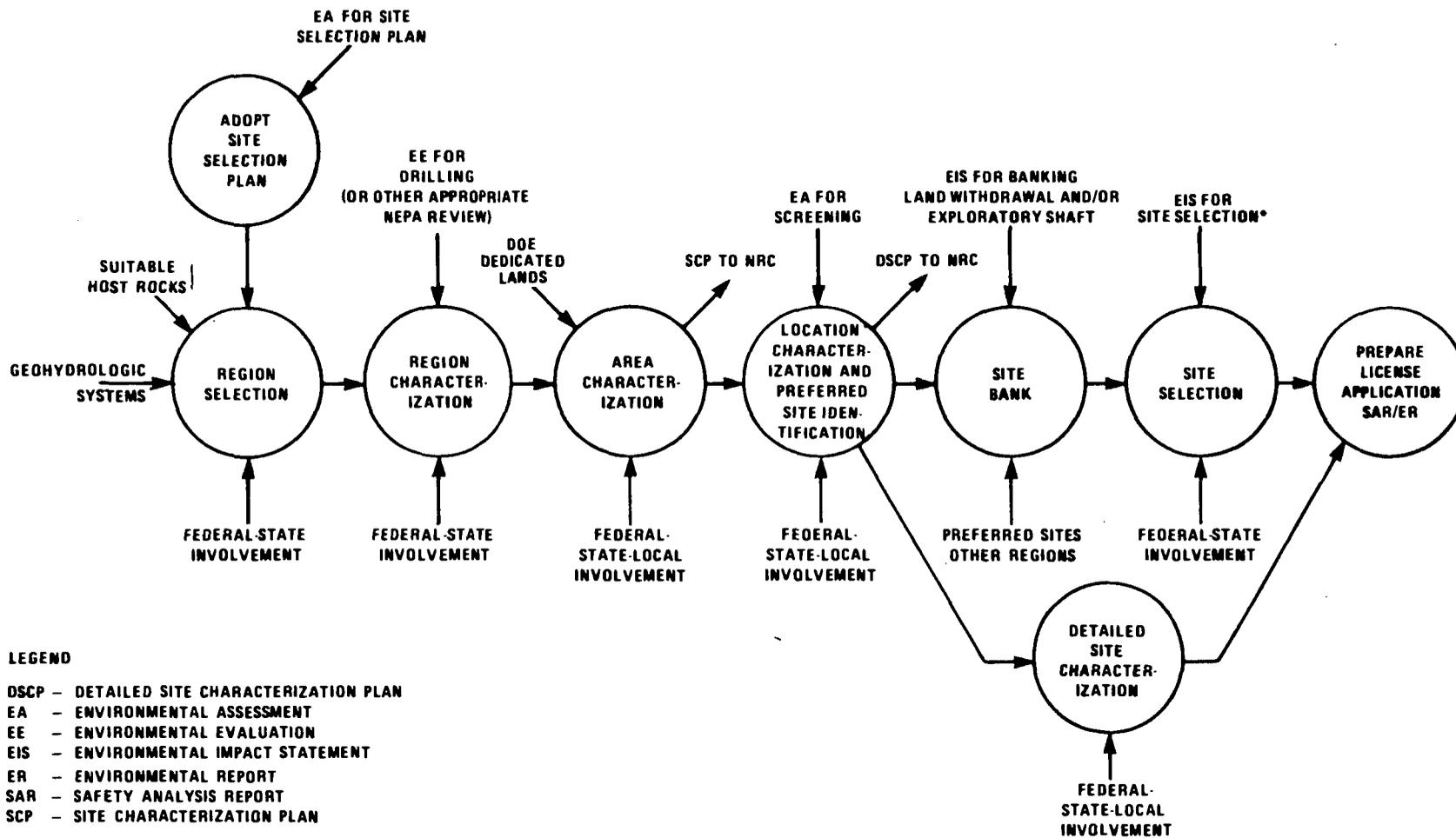


FIGURE 2.3.1. Site Characterization and Selection Process

The second major decision process is that involving the selection of sites for the disposal of nuclear waste assuming the mined geologic option. The major decision points in such a site-selection process are:

1. Adoption of a National Site Selection and Characterization Plan including the national screening for potential regions and selection of areas (approximately 2,590 square kilometers, or 1,000 square miles) for further study.
2. Identification of locations (26 to 78 square kilometers, or 10 to 30 square miles) for in-depth study.
3. Selection of a preferred site(s) for banking,^(a) including the possible development of an early shaft.
4. Acquiring an interest in land sufficient to protect potential sites from other uses.
5. Selection of a candidate site to propose to NRC for licensing as the first repository.

At each of these decision points, the DOE will consider the appropriate NEPA documentation. While the appropriate NEPA documentation is being prepared for the various decision points, program activities, including site characterization activities, that have been analyzed in previous NEPA documents may continue. In addition, further site characterization activities may continue if it is clear, based on the DOE's review, that they do not 1) have significant adverse environmental impact or 2) limit the choice of reasonable alternatives (40 CFR 1506.1). These activities could include environmental studies, routine geophysical studies, shallow drilling, and borehole drilling.

2.3.1 The DOE's National Environmental Policy Act Implementation Plan^(b)

2.3.1.1 Program Strategy

The environmental effects of implementing a program strategy are addressed in this final EIS on Management of Commercially Generated Radioactive Waste. Based upon the analyses of nine disposal concepts, mined geologic disposal is identified as the preferred technical alternative and the proposed action is the selection of a program strategy emphasizing geologic disposal in a mined repository.

2.3.1.2 Site Selection Process

National Site Characterization and Selection Plan

The DOE proposes to adopt formally the current National Waste Terminal Storage Site Characterization and Selection Plan as the comprehensive National Site Characterization and

(a) Protecting a potential repository site(s) from conflicting uses until such time as a final site(s) is selected.

(b) Section 5.2 and Appendix B.7 discuss the technical considerations of repository site selection.

Selection Plan. The current plan, described elsewhere (DOE 1980c), will be followed pending adoption of a formal plan. An EA is being prepared as input to the decision on whether to adopt or modify this plan.

The proposed plan includes:

- The methodology for identifying geographic regions for site studies.
- The methodology and criteria for screening these regions for areas, locations, and candidate sites to be studied in detail.

The environmental impacts of the methodology and criteria in the proposed plan and their reasonable alternatives will be assessed. In addition, the selection of areas for further study and the anticipated range of site characterization activities, including the environmental impacts of typical surface and subsurface activities in several environmental settings, will be analyzed. Similarly, the criteria proposed to be used to qualify and disqualify sites will be discussed.

It is believed that an EA, and not an EIS, is the appropriate level of NEPA review, since it is unclear that the decision will result in significant environmental impacts. However, upon completion of the EA, a decision will be made regarding the need to prepare an EIS. The Department of Energy will consider the results of the NEPA review prior to deciding whether to adopt or modify the proposed plan. The adopted site characterization process will be repeated in diverse geologic environments and different host media until four to five sites have been qualified.

Identification of Locations

Following completion of area studies for a particular region, in accordance with the National Plan, an EA will be prepared as input for a decision to narrow the investigations to a limited number of locations. The site-selection process to date will be described, and the environmental factors pertinent to the proposal to limit more comprehensive exploratory activities to the preferred locations will be analyzed. A comparison of environmental factors for preferred and alternate locations, based on data commensurate with the level of site-specific information available, will be provided and the environmental impacts of the range of potential exploratory activities anticipated in the location studies will be considered.

Here, too, it is believed that an EA is the appropriate level of NEPA review, since it is unclear that this decision will have environmental significance. Upon completion of the EA, a decision will be made regarding the need to prepare an EIS.

Identifying Preferred Sites for Banking/Early Shaft

At the conclusion of the location studies, the DOE will propose one or more of the sites in a location as a preferred site to be banked. Because a banked site ultimately may become the location of a repository, it is appropriate to prepare an EIS prior to the decision to bank the preferred site(s). This EIS also would provide input to a decision to acquire an interest in the site(s), if necessary, in order to maintain the integrity of the site through the site-selection process.

Using a general conceptual design for the appropriate media (a site-specific design will not be developed until after the candidate site is selected), the EIS will evaluate the potential environmental impacts of 1) a conceptual repository at the alternate sites within the region and 2) the detailed site characterization activities which may be required at each of the alternate site(s), including the possible construction of an early shaft, if required.

Although the general conceptual design will not be site-specific, it will be in an advanced stage of development relative to the medium in which the potential candidate sites are located. This will allow adequate analysis of the potential environmental impacts associated with a conceptual repository at each of the alternative sites. In addition, the interaction of waste package options with the geologic medium will be assessed in each site-banking EIS.

Site Selection

Following the banking of sites in several media, a site will be selected for a license application for the first repository. The EISs previously prepared for site banking will be supplemented, as appropriate, in an integrated EIS, which will provide a comparative environmental analysis of the alternative sites. This EIS will incorporate by reference the site-banking EISs and include any significant new information obtained since the preparation of the earlier EISs. The site-selection EIS also will serve as input to the environmental report submitted to NRC with the license application.

2.3.1.3 Land Acquisition

After a site-selection decision, the DOE may take steps to permanently acquire the site. The site banking EISs, as supplemented in the site-selection EIS, will be used as input to the land acquisition decision.

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CHAPTER 3

DESCRIPTION OF PROGRAM ALTERNATIVES AND BACKGROUND

This section describes the major action proposed by the Department of Energy for which this environmental impact statement was prepared, namely the selection of a programmatic strategy emphasizing geologic disposal in a mined repository as the technology for disposal of high-level radioactive wastes. Two programmatic alternatives to this proposed action are also described. In addition, this section provides the reader with a description of the technical and environmental bases for the analyses which follow in succeeding sections. Since radiation exposure is a central concern in the management and disposal of nuclear wastes, background information about radiation and the approaches used to assess radiological risk are presented. Finally, "non technical" issues are discussed to inform the reader about the broad social, political, and institutional concerns which cut across specific technical concerns about nuclear waste.

3.1 PROPOSED ACTION AND PROGRAM ALTERNATIVES

As part of its responsibility for developing the technology required for managing certain classes of radioactive wastes, the Department of Energy proposes to take a major agency action: selecting an appropriate programmatic strategy leading to the disposal of commercial radioactive waste in a fashion that provides reasonable assurance of safe, permanent isolation of these materials.

This major action involves two specific components at this time. The first is the selection of geologic disposal in a mined repository as the technology for emphasis in a research and development program from among the various concepts that have been proposed. The second decision concerns the nature and extent of the research and development program to be undertaken, given the designation of geologic disposal as the technology for emphasis.

In considering alternative methods that might be employed for permanent isolation of radioactive materials, this EIS identifies and examines nine disposal technologies. These technologies, fully characterized in Chapters 5 and 6, are:

- 1) geologic disposal using conventional mining techniques
- 2) disposal in very deep holes
- 3) disposal in a mined cavity that results in rock melting
- 4) disposal in repositories located on an island
- 5) disposal in sediments beneath the deep ocean in the subseabed
- 6) disposal in an ice sheet in the Arctic or Antarctic
- 7) disposal in an injection well
- 8) disposal by partitioning of reprocessed waste and transmutation of actinides
- 9) disposal by projection into outer space.

In considering the nine disposal technology concepts, a variety of nuclear wastes is considered. Each concept needs to be evaluated in terms of capability to handle both spent fuel (as a waste) and waste from fuel reprocessing. Further, the ability of these technologies to accommodate transuranic (TRU) wastes is evaluated (see Section 6.2). As shown in Table 3.1.1, not all of the technologies are capable of handling all three categories of waste efficiently. Nonetheless, some of these technologies may be useful for special purposes such as the disposal of very long-lived radioactive substances. Some concepts are rated impractical because of special handling requirements, anticipated cost, environmental risks and current capabilities to implement the technology.

TABLE 3.1.1 Potential Ability of Technology to Handle Waste Type

<u>Technology</u>	<u>Unprocessed Spent Fuel</u>	<u>High-Level Reprocessing Waste</u>	<u>TRU Waste</u>
Geologic	Yes	Yes	Yes
Very Deep Holes	Yes	Yes	I
Rock Melting	No	Yes	No
Island	Yes	Yes	Yes
Subseabed	Yes	Yes	I
Ice Sheet	Yes	Yes	I
Injection Well	No	Yes	No
Transmutation	No	Yes	No
Space	I	Yes	I

LEGEND: Yes--Concept applies
 No--Concept will not work
 I--Concept impractical.

Evaluation of these various technical alternatives for waste isolation has resulted in a finding that geologic disposal (placement of radioactive wastes in geologic formations using conventional mining techniques) is the preferred technology for research and development. However, the evaluation of these alternatives has led to the conclusion that two other disposal concepts deserve further examination as potential backup or ancillary technologies to geologic disposal: subseabed disposal (placement of wastes in sediments beneath the deep oceans), and very deep hole disposal (placement of wastes into very deep drill holes).

This Statement examines the ultimate environmental impacts of the Department of Energy's proposed action, a research, development and demonstration program emphasizing mined geologic repositories, as well as two alternative courses of action: 1) parallel development of several technologies to an approximately equal level prior to a decision on implementation and 2) the alternative of no action.

The Interagency Review Group (IRG) on Nuclear Waste Management in its report of March 1979, identified a number of alternative technical strategies, the environmental impacts of which are encompassed in the analyses contained in this Statement. The IRG Report recommended after considerable study and public input that:

- The approach to permanent disposal of nuclear waste should proceed in a stepwise basis in a technically conservative manner.
- Near-term program activities should be predicated on the tentative assumption that the first disposal facilities will be mined repositories, though nearer-term alternative approaches--subseabed and very deep hole disposal--should be given funding support.
- A number of potential sites in a variety of geologic environments should be identified, and action taken to reserve the option to use them if needed. Within technical constraints, actions should be taken to have several repositories operational before the end of the century in different regions of the country.

Beyond these recommendations, the IRG defined four alternative strategies for the development of repositories:

1. Strategy I provides that only mined repositories be considered for the first several repositories and that only geological environments with salt as the emplacement media would be considered for the first several repositories. As a result of past programs, a large body of information about salt as an emplacement medium exists. Thus, salt would be a probable choice for these repositories, since the speed of implementation of this strategy would likely rule out other media.
2. Strategy II is similar to the first, except that a choice of site for the first repository would be made from among whatever types of environments have been adequately characterized at the time of choice. However the first choice would still likely be from environments based on salt geology.
3. Strategy III provides that, for the first facility only mined repositories would be considered. However, three to five geological environments possessing a wide variety of emplacement media would be examined before a selection was made. Other technological options would be contenders as soon as they had been shown to be technologically sound and economically feasible.
4. Strategy IV provides that the choice of technical option and, if appropriate, geological environment be made only after information about a number of environments and other technical options has been obtained.

These strategies are associated with different amounts of time needed to achieve an operational repository, with Strategy I requiring the least amount of time and Strategy IV requiring the most time.

DOE, on the basis of the input from many sources, has formulated a proposed research, development and construction program for mined geologic repositories that incorporates the

recommendations of the IRG Report. Environmental impacts that would be associated with each of these differing strategies and with differences in timing of implementation (i.e., immediate versus delay) are well within the envelope of the analyses reported in this Statement. Environmental consequences associated with Strategies I through III are bounded by the environmental analyses of the Proposed Action, while those associated with Strategy IV are within the envelope of analyses performed for the Parallel Development Alternative Action. This latter action also envelopes the environmental consequences associated with a "delayed action" strategy, i.e., delaying siting of a repository until enough is known about several technical alternatives. These analyses examine the environmental consequences of constructing, operating and decommissioning waste management facilities.

3.1.1 Proposed Action

The proposed research and development program for waste management will emphasize use of mined repositories in geologic formations capable of accepting radioactive wastes from either the once-through or reprocessing cycles. This program will be carried forward to identify specific locations for the construction of mined repositories. The rationale for the selection of mined repositories as the preferred concept is presented in Section 6.2.5.

Initially, site characterization programs will be conducted to identify qualified sites in a variety of potential host rock and geohydrologic settings. As qualified sites are identified by the R&D program, actions will be taken to reserve the option to use the sites, if necessary, at an appropriate time in the future. Supporting this site characterization and qualification program will be research and development efforts to produce techniques and equipment to support the placement of wastes in mined geologic repositories.

The Department of Energy proposes that the development of geologic repositories will proceed in a careful step-by-step fashion. Experience and information gained in each phase of the development program will be reviewed and evaluated to determine if there is sufficient knowledge to proceed to the next stage of development and research. The Department plans to proceed on a technically conservative basis allowing for ready retrievability of the emplaced waste for some initial period of time.

The proposed timing for emplacement of waste into geologic repositories calls for at least two operational facilities before the end of the century. This schedule reflects the need to expand the technical evaluation of a broader set of geologic media and multiple sites and to consider a possible regional approach to repository siting. Changes in timing for emplacement of wastes in geologic repositories because of environmental or other considerations is considered within the scope of the proposed action presented in this Statement.

Some support would be provided to further evaluate the alternatives of placement in deep ocean sediments and in very deep holes. The purpose of this support is to permit continued evaluation of these technology options as alternatives to geologic disposal. These options are considered as backups or complements to geologic disposal and are presently not planned for full development.

3.1.2 Alternative Action--Parallel Development

As an alternative to emphasis on geologic disposal, the research and development program would emphasize the parallel development of several disposal technologies. This action implies an R&D program to bring the knowledge regarding two or three disposal concepts and their development status to an approximately equal level. At some later point, a preferred technology would be selected for construction of facilities for radioactive waste disposal.

Based upon the Department of Energy's evaluation, the likely candidate technologies for this parallel development strategy would be:

- 1) geologic disposal using conventional mining techniques
- 2) placement in sediment beneath the deep ocean (subseabed)
- 3) disposal in very deep holes.

In order to develop several technologies in parallel, the range of approaches within each disposal technology would likely be narrowed to a single candidate approach.

The geologic disposal program would concentrate on a most preferred geohydrological system and, possibly, host rock. By narrowing the focus of the program, resources of time, money, and manpower would be made available to pursue the parallel development programs of the other two technologies.

In a similar fashion, the subseabed program would focus on a preferred system for waste emplacement and on a few locations.

The program activities for very deep hole disposal would eventually be focused on specific deep geohydrological systems and in specific regions of the country. Since adequate information about such deep systems is not currently available to do this, a program of study would need to be developed to acquire such information.

The strategy to develop several disposal technologies in parallel requires the use of extended term storage facilities since significant additional time would be required to bring the technologies of sub-seabed and very deep hole disposal to a level of development equivalent to that of geologic disposal. The main differences between the Proposed Action and the First Alternative Action are the degree of emphasis on geologic disposal and the timing of actual construction of waste disposal facilities.

3.1.3 No-Action Alternative

This alternative would eliminate or significantly reduce the Department of Energy's research and development programs for radioactive waste disposal. Under this alternative, existing spent fuel would be left indefinitely where it is currently stored and any additional spent fuel discharged from future operation of commercial nuclear power plants would likewise be stored indefinitely in water basin facilities either at the reactors or

at independent sites. The Department of Energy does not consider this no-action alternative to be a reasonable course, since it offers no solution for the long-term period beyond the useful life of the water basins.

3.2 BASES FOR THE ANALYSIS

A number of bases for analysis must be established to assess environmental impacts associated with a nuclear waste disposal technology. This includes the identification and description of predisposal facilities necessary for waste management, as well as a description of the disposal facilities themselves. Further, the physical, biological and social environments into which these facilities will be placed must be characterized. However, total or net environmental impacts cannot be described completely by the effects of single facilities in the environment, so this Statement also analyzes complete waste management systems. The key assumptions associated with a systems analysis are those of nuclear power growth (i.e., amount of waste to be disposed) and the nuclear fuel cycles considered (i.e., kinds of waste to be disposed).

The general approach to environmental assessment used here investigates potential impacts associated with construction, operation (including potential accidents), and decommissioning of predisposal facilities (including treatment, transportation and storage of wastes) and the repository system itself. Physical protection requirements for safeguarding the wastes from theft or sabotage are also evaluated. Impacts resulting from nuclear waste disposal include those associated with resource commitments, ecological and atmospheric effects, radiological effects, socioeconomic effects, and the costs of waste management and disposal.

Predisposal facilities are discussed in Chapter 4, and geologic repositories are discussed in Chapter 5. Conceptual facilities are described, their impacts and costs of construction and operation are estimated, and safeguard requirements are evaluated. These conceptual facilities and impacts are described in detail in Technology for Commercial Radioactive Waste Management, DOE/ET-0028, April 1979 and Environmental Aspects of Commercial Radioactive Waste Management, DOE/ET-0029, April 1979. Summary descriptions and key results are presented in Chapters 4 and 5.

A description of the physical environments for the different facilities is given in Chapter 5 for geologic disposal and in Chapter 6 for alternative technologies. The biological and social environments used hypothetical or reference conditions which were assumed common to all geologic repositories and associated waste management facilities. For assessing general environmental and health effects for these facilities, a single reference environment was developed and is described in Appendix F. This reference environment provides the necessary description of environmental characteristics (e.g., demography, atmospheric dispersion patterns, surface waters, plant and animal communities) that serve as a baseline for generically estimating environmental impacts of waste management and disposal. Three reference environments were used to assess the socioeconomic impacts of the influx of workers associated with geologic repositories and related facilities, because socioeconomic impacts are particularly sensitive to variation in demography (Appendix G). The use of reference environments should not be construed as an endorsement of particular regions for siting waste management and disposal facilities but rather as convenient and realistic assessment tools. Different reference environments and bases for analyses were used in the case of alternative disposal technologies and are described where used in Section 6.1.

In Chapter 6, alternatives to geological disposal in mined continental repositories are described, evaluated, and compared.

In Chapter 7, the requirements and impacts for entire waste management systems for several different nuclear industry growth assumptions are described. These requirement and impact descriptions incorporate information about the individual waste management components (described in Chapters 4 and 5) into system simulation calculations.

The assumptions used regarding nuclear fuel cycles and industry growth as well as the basis for assessing resource commitments, ecological and atmospheric effects, radiological effects, socioeconomic impacts, potential accidents, physical protection, and costs of management and disposal of nuclear wastes are described in the following subsections.

3.2.1 Nuclear Fuel Cycle Assumptions

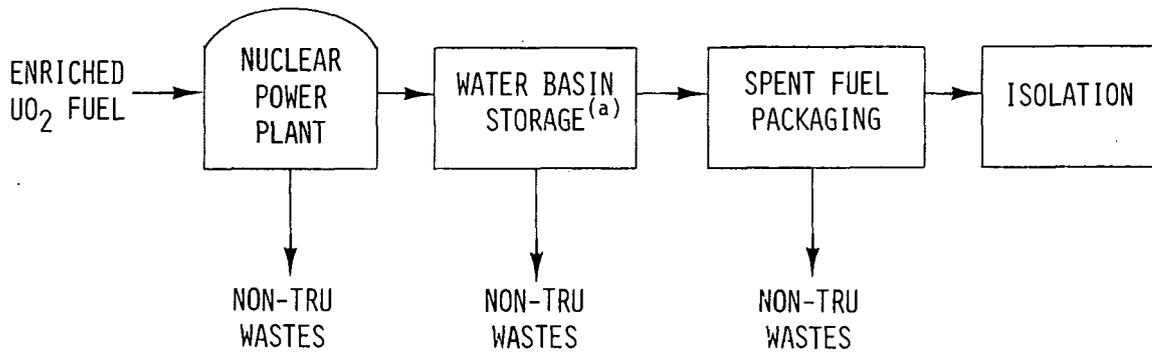
The waste management impacts of two basic light water reactor (LWR) fuel cycles are analyzed in this Statement. These are 1) the once-through fuel cycle where spent fuel is sent to disposal without reprocessing for recovery of residual energy potential, and 2) the reprocessing fuel cycle where spent fuel is determined to be a resource and is processed for recovery and use of the contained uranium and plutonium. A uranium-only recycle case (with plutonium remaining in the high-level waste or recovered and stored elsewhere) was considered in the draft of this Statement. However, because of the low likelihood that this fuel cycle would ever be implemented and because of comments to this effect received on the draft Statement, it has been deleted from this final Statement. Information on this fuel cycle may be found in DOE/ET-0028 and DOE/ET-0029.

3.2.1.1 Once-Through Fuel Cycle

A simplified diagram presenting the once-through cycle is shown in Figure 3.2.1. Spent fuel is stored until a qualified Federal waste isolation facility is in operation. Storage can occur either at the reactor site or at an offsite away-from-reactor (AFR) storage facility, also sometimes referred to as an independent spent fuel storage facility (ISFSF). Storage at an AFR is necessary if sufficient storage capacity is not available at nuclear power plant sites. At the AFR, only nontransuranic and gaseous wastes are generated^(a) while the spent fuel is handled and stored. Thus, the only waste of concern to this Statement is the spent fuel itself. The following assumptions are made about the once-through fuel cycle.

- Although storage capacity in the nuclear power plant (reactor) basins will vary considerably and may be increased significantly for new plants, a given reactor basin will have, on the average, the capacity for seven annual discharges in addition to full core reserve. This capacity assumption results in away-from-reactor

(a) Strictly speaking, the radioactivity content in the wastes is "generated" during irradiation of the fuel in the nuclear power plant.



(a) WATER BASIN STORAGE IN EITHER REACTOR BASINS OR AFR FACILITIES

FIGURE 3.2.1. Once-Through Cycle

storage requirements that approximate the maximum requirements shown in a recent study when currently licensed expansion plans are all assumed to be implemented and full core reserve capacity is maintained (DOE/NE-0002 1980). Implications of variations in reactor storage capacity are discussed in the Final Environmental Impact Statement on U.S. Spent Fuel Policy (DOE/ET-0015 1980).

- To permit the spent fuel to cool down prior to dry encapsulation and disposal the spent fuel is stored for a minimum of 5 years in the nuclear power plant storage basins for the reference once-through fuel cycle. If a disposal facility is not available, the spent fuel remains stored at the reactor until the 7-yr capacity is filled, after which excess fuel older than 5 years is shipped (Section 4.5) to an AFR (Section 4.4) where it remains until a disposal facility is available.
- Spent fuel encapsulation (or packaging) facilities (Section 4.3) are located on the same site as the disposal facility. An alternative of encapsulating the spent fuel at the AFR and storing packaged spent fuel is also described in the predisposal system discussions in Sections 4.3 and 4.4.
- For purposes of estimating transportation impacts, shipping distances from reactors to an AFR average 1000 miles for this generic statement. Shipping distances from reactors to a repository or from an AFR to a repository are assumed to average 1500 miles. Therefore, total shipping distance between a reactor and disposal can be as much as 2500 miles. Actual shipping distances would vary, of course, depending on sites selected.

The logistics and storage requirements of this fuel cycle for several nuclear power growth assumptions are discussed in Chapter 7.

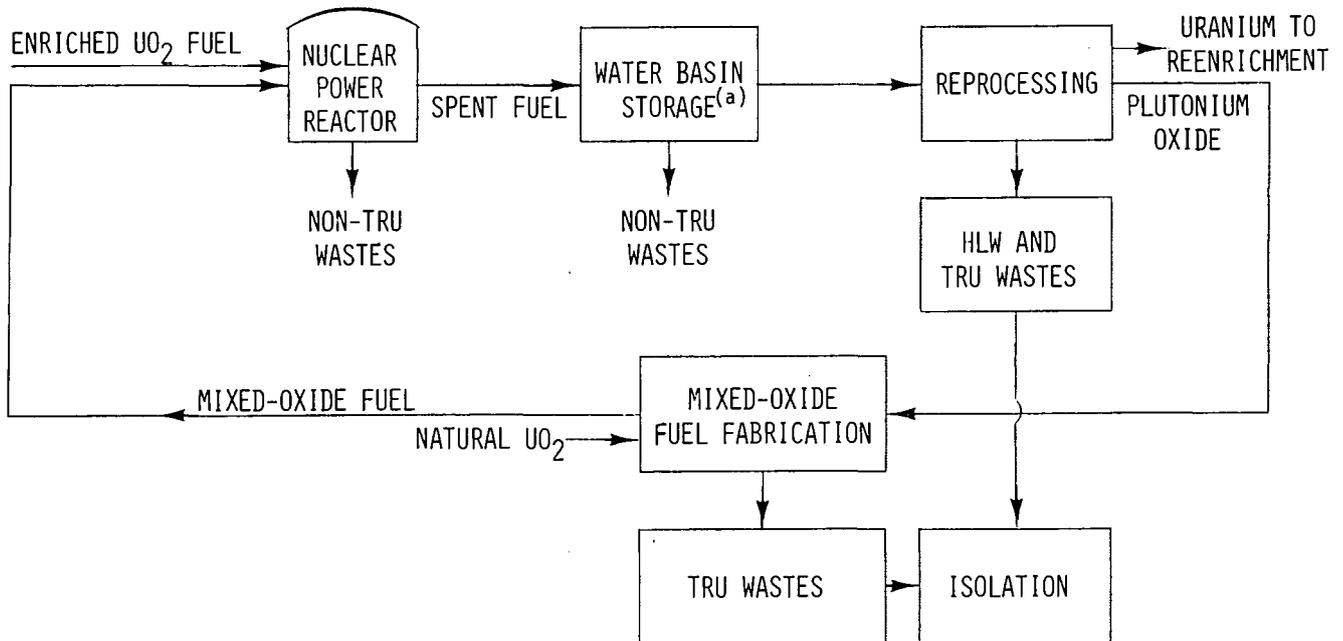
3.2.1.2 Reprocessing Fuel Cycle

A simplified diagram of the reprocessing fuel cycle is shown in Figure 3.2.2. In this fuel cycle, uranium and plutonium are separated from other components of the fuel and

purified for recycle at a fuel reprocessing plant (FRP). The major process steps at the FRP, excluding waste treatment operations, which are described in Chapter 4, are:

- Underwater storage of spent fuel awaiting processing.
- Recovery and purification of the uranium and plutonium by solvent extraction using the Purex process. The reference plant, described in DOE/ET-0028, Section 3.2, operates 300 days per year to process 2000 MTHM/yr of spent fuel. The spent fuel elements are chopped into short sections so that the contained fuel can be dissolved in nitric acid. The uranium and plutonium are then extracted into an organic solvent phase containing tributyl phosphate (TBP), leaving the bulk of the fission products in the nitric acid solution (the high-level waste). The uranium and plutonium are separated and the remaining fission products removed in subsequent solvent-extraction process cycles.
- Conversion of plutonium to a solid at the FRP by precipitating plutonium as an oxalate, which is then separated and calcined to PuO_2 .
- Conversion of the uranium from a nitrate solution to UF_6 at the FRP by calcining the uranium nitrate to UO_3 , reducing the UO_3 to UO_2 with hydrogen, then converting the UO_2 to UF_4 by hydrofluorination with HF, and finally converting the UF_4 to UF_6 with fluorine. (UF_6 is the form required by the enrichment plant.)

Over 99% of the spent fuel fission products and about 0.5% of the uranium and plutonium would be contained in the FRP high-level waste. Substantial quantities of a variety of TRU



(a) WATER BASIN STORAGE IN EITHER REACTOR BASINS, AFR FACILITIES OR FRP BASINS

FIGURE 3.2.2 Uranium-Plutonium Recycle Fuel Cycle

wastes also result. These are described more fully in Section 4.2. After the HLW is solidified (Section 4.3) it may be stored on-site (Section 4.4) for a period prior to shipment (Section 4.5).

A mixed-oxide fuel fabrication plant (MOX-FFP) prepares fuel containing a mixture of plutonium dioxide and uranium dioxide for recycle to a nuclear power plant. The reference MOX-FFP receives UO_2 and PuO_2 powders and Zircaloy cladding tubes and end plugs and prepares hermetically sealed fuel rods ready for insertion into fuel assemblies. The reference plant, described in DOE/ET-0028 Section 3.2, operates 300 days per year to produce 400 MTHM of LWR fuel/yr; up to 5% of the heavy metal content is plutonium. The major process steps involved include:

- Mechanical mixing of UO_2 and PuO_2 powders
- Preparation of dense fuel pellets by pressing, sintering, and grinding the mixed powder
- Sealing the pellets in Zircaloy cladding to form fuel elements
- Scrap recycle. The following assumptions are made about the reprocessing fuel cycle logistics:
 - Spent fuel is stored until it is shipped to a reprocessing facility. As in the once-through cycle, storage can occur either at the reactor site or at an AFR. Reactor basin storage capacity is also seven annual discharges, but spent fuel is stored for a minimum of one year, once this accumulated backlog of stored fuel is worked off. The reprocessing plant maintains a working inventory of 0.5-yr worth of spent fuel in storage. Thus, the minimum fuel age at reprocessing is 1.5 years; however, because a large accumulated inventory of spent fuel exists before the start of reprocessing, it is over 20 years after reprocessing starts before this minimum age is reached.
 - The high-level waste is solidified immediately and then stored on-site for 5 years prior to shipment to a repository or to an interim storage facility if a repository is not available.
 - TRU wastes are shipped immediately after treatment and packaging to either a repository or interim storage.
 - Spent fuel shipping distances are assumed to average 1000 miles from reactors to an FRP or to an AFR, or from an AFR to an FRP.
 - Treated waste shipping distances are assumed to average 1000 miles to interim storage and 1500 miles from either an FRP or from an interim storage facility to a repository. As in the once-through cycle, the actual distances will vary. No waste shipments between an FRP and a MOX-FFP are assumed.

The logistical and storage requirements of this fuel cycle as well as the once-through cycle for several nuclear power growth assumptions are discussed in Chapter 7.

3.2.2 Nuclear Power Growth Assumptions

To cover the range of potential waste management impacts in the years ahead, five different nuclear power growth scenarios are considered in this Statement.

A reference projection of 400 GWe of installed nuclear power capacity in the year 2000 and a bounding low projection of 255 GWe in the year 2000 was used in the original draft Statement (DOE/EIS-0046 D). Since that report was published for comments, however, studies (Clark and Reynolds 1979) conducted by DOE's Energy Information Administration (EIA) have indicated that the year 2000 installed nuclear power capacity is unlikely to exceed 250 GWe.^(a) In addition, some comments on the draft Statement stated that the 400 GWe projection indicated a bias in favor of nuclear power development while other commenters objected that it overstated the magnitude of the waste management problem. For these reasons, the maximum projection for the year 2000 considered in this final Statement has been established as 250 GWe.

None of the projections or scenarios are intended to represent predictions of future developments. They are intended to encompass a possible range of nuclear power development and to provide a reasonable basis for estimates of waste management impacts as well as a basis for either interpolating waste management impacts to intermediate projections or for extrapolating waste management impacts to higher projected growth rates.

The waste management impacts for these scenarios are presented in Chapter 7.

The five scenarios are described below and the resulting nuclear power capacities are tabulated in Table 3.2.1 and plotted in Figure 3.2.3.

TABLE 3.2.1. Nuclear Power Capacity Assumptions, GWe

	<u>Case 1</u> <u>Present</u> <u>Inventory</u>	<u>Case 2</u> <u>Present</u> <u>Capacity</u>	<u>Case 3</u> <u>250 GWe</u> <u>in 2000 and</u> <u>Phaseout</u>	<u>Case 4</u> <u>250 GWe</u> <u>in 2000 and</u> <u>Constant</u>	<u>Case 5</u> <u>250 GWe</u> <u>in 2000 to</u> <u>500 GWe</u> <u>in 2040</u>
1980	50	50	55	55	55
1985	0	50	113	113	113
1990	0	50	155	155	155
1995	0	50	196	196	196
2000	0	50	250	250	250
2005	0	49	249	250	281
2010	0	44	244	250	312
2015	0	14	214	250	343
2020	0	0	195	250	374
2025	0	0	137	250	405
2030	0	0	95	250	437
2035	0	0	54	250	468
2040	0	0	0	250	500

(a) The referenced report did not project beyond 1995. The figure of 250 GWe in the year 2000 is based on an extrapolation.

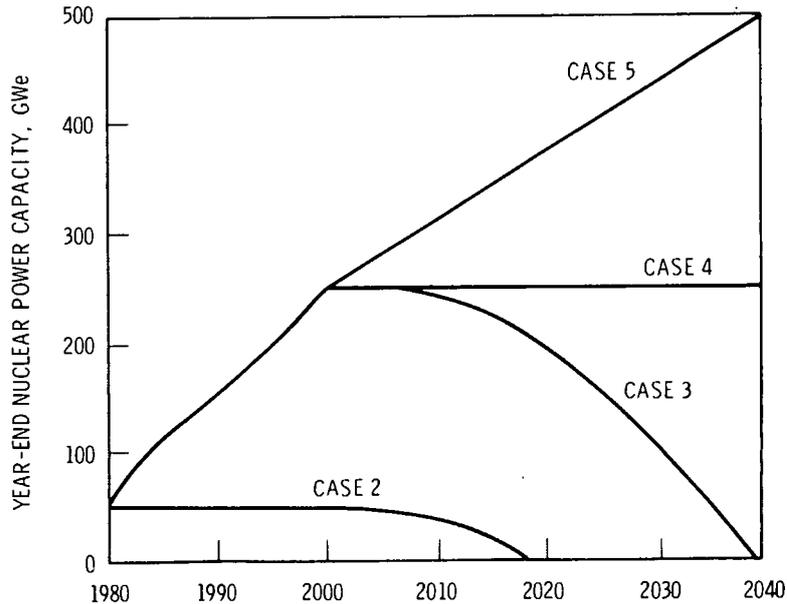


FIGURE 3.2.3. Nuclear Power Growth Assumptions

Case 1--Present Inventory--This case considers the requirements for management of approximately 10,000 MTHM of spent fuel that would remain if the 50 GWe of LWR capacity operating at the beginning of 1980 were shut down at the end of 1980 and all reactor cores discharged. However, no attempt is made in this Statement to consider or evaluate the broader issues of an industry shutdown (beyond those associated with handling the waste) such as national energy policy, impact on the economy, the impacts of alternative energy sources, costs, and the environmental impacts of such action.

Case 2--Present Capacity--This case considers the requirements for management of 48,000 MTHM of spent fuel that would result from continued operation of the existing 50 GWe of nuclear capacity to retirement after 40 years of operation with no further additions to this system. As in Case 1, no attempt is made to consider or evaluate the broader issues beyond the impact of handling the associated wastes, that would be involved in a limitation of this sort.

Case 3--250 GWe in Year 2000 and Phaseout--Case 3 assesses the waste management impacts for all aspects of a complete life cycle of a nuclear generating system including reactor shutdown, facility decommissioning, etc. In this case nuclear power capacity increases to 250 GWe in the year 2000. (This case follows the EIA high case projection through 1995.) After the year 2000, no additional nuclear power plant startups are considered. All nuclear power plants are assumed to operate for a 40-year life, after which they are decommissioned. Thus, the installed generating capacity of the system is reduced to zero in the year 2040. Based on average experience to date, average startup capacity factors of 59%, 63%, and 67% were assumed for the first three years of operation for all nuclear plants. Starting with the fourth year, each plant was assumed to operate at 70% for 22 years and then decline to 40% in its fortieth year after which it is shut down. A total of 239,000 MTHM of spent fuel is produced in this case.

We do not yet have sufficient operating experience with nuclear plants to predict this life cycle with high confidence. These plants are generally assumed to have lifetimes in the range of 30 to 40 years. The upper end of this range was used here to be conservative in regard to the amount of radioactive waste to be managed for a specific system. The declining load factor as facilities age has not yet been observed in nuclear plants but is similar to the experience of large central-station fossil-fuel generating units.

Using the year 2000 as a reference point, the impacts of other growth assumptions can be derived by comparison to this case. For example, a 500 GWe system in the year 2000 would produce approximately twice the impacts of Case 3 if allowed to run out its useful life, or a 125 GWe system in the year 2000 would produce approximately one-half as much impact.

Case 4--250 GWe in Year 2000 and Constant--This case follows the same growth pattern as Case 3 up to the year 2000. Then, instead of phasing out capacity as plants are decommissioned, new capacity is added to maintain the total capacity at 250 Gwe until the year 2040, beyond which time the case is not analyzed. A total of 316,000 MTHM of spent fuel is produced in this case.

This case illustrates the rate at which continuous waste management requirements and impacts would occur in a constant or steady-state system. An approximate equilibrium is established.

Waste management requirements and impacts at other constant capacity levels can be obtained by comparing capacities and impacts to this case.

Case 5--250 GWe in Year 2000 and 500 GWe in 2040--This case also follows the same growth pattern as Case 3 up to the year 2000. After that, however, capacity additions continue until a doubled capacity of 500 GWe is reached in the year 2040. Beyond the year 2040, the case is not analyzed. A total of 427,000 MTHM of spent fuel is produced in this case.

No equilibrium is established in this case. It illustrates the waste management requirements and impacts for a continuously expanding system. Results can be extrapolated to other growth rates by comparing the differences between the year 2040 capacities in Cases 4 and 5 to the difference in impacts. For example, a capacity of 750 GWe in the year 2040 would have twice the additional impact over Case 4 that Case 5 has.

3.2.3 Resource Commitment Assessment

In most instances, data describing environmental impacts that are caused by commitments of resources are presented as land and water requirements, material requirements, energy consumption, and manpower requirements for construction, operation, and decommissioning of the facilities. Resource commitments are combined by facilities on a single reference plant basis for analyzing predisposal activities in Section 4.7 and for geologic repositories in Section 5.4. Resource commitments are further aggregated by plant to systems of waste management and disposal within fuel cycle options in Chapter 7.

3.2.4 Ecological and Atmospheric Impacts

The impacts of the treatment, interim storage, transportation, and final disposal of radioactive wastes on natural ecosystems cannot be satisfactorily dealt with in detail in a generic sense because of the overriding influence of site-specific factors. For example, the expected impacts of certain waste technologies on plant and animal communities in an area of high precipitation may be markedly different from those in an arid environment. The ability of natural systems to withstand stress will vary widely according to their environment. Similarly, the economic worth of the natural resources at risk will depend greatly on the region and the degree of change already induced by human activities.

In this Statement, the assumption is made that environmental releases of radioactive wastes that are within the acceptable standards designed to protect man will also be within limits tolerable to natural plant and animal populations. In general, man is believed to be more sensitive to radiation than are other lifeforms. Thus, the discussion of potential radiation effects on plants and animals other than man is not considered on a generic basis. Consequently, discussion of the ecological impacts of radioactive waste management is confined mainly to 1) the effects on the use of land and surface water and 2) the impacts resulting from the release of nonradioactive chemicals and heat to the air and to surface water.

The main atmospheric effects evaluated in this Statement are the impacts on ambient air quality caused by emissions to the atmosphere during construction and operation of the facilities. Secondary emissions from construction force vehicles and construction equipment are also included in the emissions inventory. Since heat is a by-product of each process, its effect on the biosphere, whether released directly or via cooling tower, is also investigated.

3.2.5 Radiological Impacts Assessments and Uncertainties

Radiological impacts are probably perceived as the most important aspect of radioactive waste management. As a consequence, radiological aspects are considered in detail in this Statement and in its supporting documents. Radiological impacts are described principally in terms of dose to workers and to the public (The regional population is described in Appendix F; mathematical models are described in Appendix D.)

Doses to the public from waste management operations would be expected to arise from inhalation of radionuclides, by direct radiation, and from ingestion of food products (e.g., vegetables, meat, and dairy products) either grown on land contaminated by radionuclides deposited on the ground or contaminated by deposits directly on the food products themselves.

Dose from exposure to planned or unplanned releases of radionuclides to the biosphere is considered for three main categories of the public: the maximum individual,^(a) the

(a) The maximum individual is a hypothetical resident whose habits would tend to maximize his dose.

population within a 50-mile radius reference environment of a waste facility (2 million), and the world population (6 billion in the year 2000).^(a) In selected instances dose to the population of the eastern half of the United States is also presented.

Unless otherwise noted, doses are to the whole body; doses to other organs of interest are presented in DOE/ET-0029. Dose in this Statement is usually expressed as a 70-yr accumulated whole-body dose, although where informative, first-year doses are also given. In some instances, multigeneration doses are provided.

Health effects are calculated for regional or worldwide populations based on the dose received by these populations from the aggregation of the facilities involved. The doses calculated to result from individual facilities, except for nondesign basis repository accidents, are usually too small to warrant discussion of health effects.

In this Statement, 50 to 500 fatal cancers and 50 to 300 serious genetic defects are assumed to result in an exposed population for each million man-rem of radiation exposure received (for a total of 100 to 800 health effects per million man-rem). The possibility of zero risk is not excluded by the available data, i.e., there is a possibility that no cancers may be caused by low doses of radiation. For further discussion of the derivation of these risk factors, the reader should consult Appendix E.

Also presented is an alternative approach to analysis of exposure in which the estimated radiation doses from waste management activities are compared with more accurately known radiation doses from other sources such as naturally occurring radiation and radioactive materials.

Radiation dose calculations (Appendix D) use models to develop total doses by summing radiation doses from various radionuclides entering (or externally exposing) the human body. Each step in the dose calculation has uncertainty associated with it. A common radiation protection practice has been to assign values to parameters used in dose calculation that, if uncertain, will tend to overstate rather than understate the resulting dose.

3.2.6 Socioeconomic Impacts

The approach used in the analysis of socioeconomic impacts emphasizes changes in local employment and population caused by the construction and operation of a waste repository in selected geologic media. The repositories examined in this analysis generate socioeconomic impacts in several ways: through the employment requirements of construction and operation, through the demand generated for locally supplied materials and services, through secondary economic growth generated by the project, and through the public revenues resulting from project operation. In this generic Statement, the employment requirements are stressed because they more directly affect impacts (such as demands for housing, education, and health services) than do other requirements. Because tax structures and prospective revenues vary widely across potential sites no meaningful and representative estimates of reve-

(a) The only radionuclides that contribute significantly to worldwide radiation doses for the type of release mechanisms visualized here are ^3H , ^{14}C , and ^{85}Kr . For this reason, worldwide dose calculations are based on ^3H , ^{14}C , and ^{85}Kr only.

nue impacts can be provided in a generic study and no such estimates are prepared in this Statement.

A baseline population from the start of construction of a facility until scheduled decommissioning is projected. Work force requirements for the project are compared with the availability of workers already living in the area. Workers not available within commuting distance of the site will immigrate. The impact of their presence in the local area is increased to the extent that they either induce secondary growth in the local economy or bring family dependents with them. The total influx of new people to an area can equal three or four times the number of primary workers hired from outside the area. The model distributes the total new population to the site county and surrounding counties on the basis of county size, distance to the work site and availability of housing.

A generic assessment of the socioeconomic impacts incorporates the assumption that a variety of sites are potential candidates. Since the potential sites may differ considerably in terms of their distinguishing characteristics (especially population size, composition and distribution, industrial composition of the labor force, and availability of social services), the potential effects of project development on a number of alternative sites must be examined. In order to emphasize that the reference sites used in this analysis are hypothetical, they are simply labeled Midwest, Southeast, and Southwest. Each reference site consists of a single county. The region within which the county is located is defined as the aggregation of all counties falling substantially within a 50-mile radius of the site. The forecasting model allocates immigrants to these counties, then focuses upon the new population residing in the site county and upon the demands it places upon the county for social services. The objective of this generic analysis is to provide a range of probable socioeconomic impacts and to illustrate how variation in site characteristics and variations in construction and operating requirements with different disposal media combine to produce demographic and economic pressures upon local areas. Whether or not these pressures become translated into actual net socioeconomic impacts depends upon how each community responds in terms of the capacity of the service system to absorb new demands, the willingness of the community to adjust to pressure for change, and the availability of mitigating strategies to the community.

3.2.7 Basis for Accident Analysis

The accident analysis procedure for this Statement involves several steps. First, potential accidents are identified for each waste management function and alternative technology. Next, accidents are divided into four categories based on considerations of their potential to expose plant workers to significant radiation levels and/or release radioactive material to the environment. Accidents in each severity category are then grouped by similar release characteristics. Finally, the largest potential accident release category/accident severity group is selected for environmental consequence analysis. In all, 207 possible accident types were examined for the waste management system with 116 of these having potential for offsite releases of radioactive material. Forty-six (46) of the releases were analyzed for environmental impacts.

A listing of all accidents considered in this analysis and the grouping of releases to determine source terms for environmental consequence analysis is given in Section 3.7 of Technology for Commercial Radioactive Waste Management (DOE/ET-0028). Environmental impacts of specific source terms are presented in the Environmental Aspects of Commercial Radioactive Waste Management (DOE/ET-0029).

Each waste management technology was examined for potential accidents which might result in offsite releases or significant impact on plant operations. Potential hazardous material releases (called source terms) were developed for these accidents using successive release fractions. The release fraction is the fraction of radionuclide inventory that is released to the next containment barrier or to the environment. The radioactivity released in an accident may be substantially reduced by one or more barriers, such as high-efficiency particulate air (HEPA) filter banks. The radioactivity released to the environment was obtained by multiplying the product of the release fraction for each release mechanism and containment barrier (e.g., the accident, process equipment, HEPA filters, etc.) by the radionuclide inventories involved in the operation. Where more than one waste management technique was examined, analysis was based on the example system waste form (see figure 4.1.3 on page 4.8 for the identification of the example waste forms).

Accident frequency estimates were developed where possible. In the absence of actual accident experience estimates are based on previous experience with similar equipment, while others are engineering judgment based on review of the conceptual designs.

Following source term and frequency definition, the lists of representative accident scenarios were classified into three accident severity groups:

1. Minor--Process interruptions without potential for significant release of radioactive or other hazardous materials.
2. Moderate--Events with potential for small radioactivity release.
3. Severe--Events with a potential for significant radiation hazards.

The three accident classifications cover the spectrum of design-basis accidents. Non-design-basis accidents (a fourth category) includes all accidents which exceed site criteria^(a) (e.g., meteorite impact) or involve concurrent independent failure of process and multiple containment system barriers. By virtue of plant design and operational techniques, the possibility of nondesign-basis accidents is extremely unlikely during the design life of the waste treatment or storage facility and are not considered for these facilities. However, for geologic isolation, because of the long period of required containment, several nondesign-basis accidents (or unexpected events) are postulated (Section 5.5).

An umbrella source term concept was used to limit the number of accidents requiring detailed impact analysis. Viewed independently of accident initiation sequences and fre-

(a) Site criteria include: 1) definition of the maximum credible earthquake, surface faulting, floods and wind velocities based on historical evidence, local and regional geology, and expert judgment; 2) local and regional demography; and 3) proximity and definition of hazards caused by man.

quencies, source terms can be grouped by release severity for environmental consequence analyses. Releases were classified based on similar release pathways, chemical form, accident severity category, and isotope types released (fission products, activation products, and actinides). The largest release from any of the accidents in a similar release group was selected as the umbrella source term for that group. A summary description of impacts from the umbrella source terms for each waste management step is presented in Sections 4.8 and 5.4.

Releases of radioactive material to the environment result from both accidents and normal operational releases. Operational releases result from routine handling or processing of radioactive materials and are limited by the containment system design and performance. They are expected to occur at a relatively uniform rate over the life of the plant. Accidental releases occur intermittently because of operational error or because of system component or containment failures. Severity of releases is generally inversely proportional to their frequency. The small-release, moderate-frequency minor accidents were characterized for impact analysis in two ways: 1) as short-term intermittent release to describe their accidental nature and 2) as integrated releases averaged over one year to describe their moderate frequencies of occurrence. Integrated annual releases caused by minor accidents were added to facility releases from normal operations in determining environmental impacts for normal operation. Because of their low frequency, releases from moderate and severe accidents are described as separate impacts and are not included in consequences of routine operation.

3.2.8 Cost Analysis Bases

Estimates of capital and operating costs for waste management predisposal operations and disposal in geologic repositories were developed for this Statement. This section summarizes the assumptions and methodology used to derive these cost estimates, as well as the bases for estimating uncertainty ranges. A complete discussion of cost bases and assumptions is given in DOE/ET-0028, Vol. 1, Section 3.8.

The cost estimates themselves are summarized in Sections 4.9 and 5.6 for predisposal and geologic-isolation operations, respectively. Additional cost information on other disposal alternatives where the data base is generally more limited, is presented in the individual discussions of these alternatives in Chapter 6. An analysis of the overall systems costs of waste management and their impact on the cost of electric power is given in Chapter 7. The costs presented in Chapter 7 represent a full cost recovery of all identifiable costs including R&D costs and government overheads.

3.2.8.1 Bases for Capital, Operating and Decommissioning Cost Estimates

A constant dollar method of analysis is employed in which all costs, both present and future, are expressed in terms of the buying power of the dollar in mid-1978.^(a) This is

(a) The costs from DOE/ET-0028 were originally derived in terms of 1976 dollars and have been escalated here to 1978 dollars by multiplying by 1.17. 1980 dollar costs can be approximated by multiplying by 1.20.

not meant to imply that inflation will not occur; rather, cost relationships can be more easily understood and placed in perspective if they are stated in constant dollar terms. Over the long term, the estimated costs developed in this study will increase at a rate comparable to the general rate of inflation.

Capital costs were derived by estimating requirements for major equipment, buildings and structures, site improvements, and construction labor. Factors were then applied to these direct cost estimates to generate other direct costs, indirect costs, architect-engineer costs, owner's staff costs during construction, initial inventory costs and other startup costs.

Operating costs include all cost items identified with operation. The number of man-hours, quantities of materials, and requirements for utilities were derived in each case from the facility descriptions. The allowances for maintenance, overhead, and miscellaneous costs were derived by applying factors to either capital or direct labor costs.

The capital and operating cost methodology outlined above is used to estimate all of the costs given in this Statement except for those of the transportation facilities (cost development for transportation is discussed separately in Subsection 3.2.8.4). An allowance for working capital is also provided. Working capital is defined as the cash required to operate a facility, i.e., the difference between current assets and current liabilities. This cash is treated as an outflow of funds during the first year of plant operation and as an inflow during the last year of operation. Working capital requirements are estimated at 50% of the first year's operating cost.

The cost of waste management in this Statement also includes the cost of facility decommissioning. Specific cost estimates were developed for decommissioning a reference spent fuel storage facility, mixed oxide fuel fabrication plant, and fuel reprocessing plant. Based on these estimates, the costs to decommission individual waste management facilities not otherwise included in the decommissioning of these primary facilities were estimated at 10% of their capital costs (except for underground repository facilities for which separate estimates were made). These costs are incorporated in the levelized unit cost calculations for these waste management facilities. The costs of decommissioning FRP and MOX-FFP facilities are included in the waste management system costs (Section 7.6).

3.2.8.2 Bases for Levelized Unit Cost Estimates

Levelized unit costs are capital and operating costs translated into equivalent, constant (or level) annual unit costs. The unit cost is sufficient to pay any interest charges on debt; pay all operating expenses, taxes and insurance; earn a specified return on outstanding capital; and recover the capital investment over the life of the project. In summary form the levelized unit cost relationship can be expressed as:

$$\text{Levelized Unit Cost} = \frac{\text{Annualized Capital and Operating Costs}}{\text{Annualized Units Processed}}$$

Since the calculated unit costs are a function of taxes and returns on equity and debt, ownership for each facility is defined as either private industry, Federal, or utility

ownership. The constant dollar weighted average cost-of-money rates and ranges (excluding an inflation premium) used in the levelized unit cost estimates are $10 \pm 4\%$, $7 \pm 3\%$ ^(a) and $7 \pm 2\%$ for private industry, Federal, and utility ownership, respectively. Also included in the unit cost calculations are property taxes and state income taxes as well as Federal income taxes, accident and hazard insurance, and investment credits.

For this Statement, most unit costs are based on a 15-yr economic plant life. The text notes when plant lives other than 15 years are used, as in some of the storage facilities. However, because of the cost-of-money effect over long time periods at the rates employed here, plant lives longer than 15 years have only a small effect on unit costs. Although it is not anticipated, the entire facility could be replaced after 15 years with no increase in unit costs (in constant dollars) beyond those estimated here.

3.2.8.3 Uncertainty Ranges for Cost Calculations

Uncertainties in the levelized unit cost estimates were derived from uncertainties calculated for three components: 1) capital costs, 2) operating costs, and 3) the cost of money. The range for capital costs reflects uncertainties in the definition of the engineering scope required to provide a fully-functional plant based on the technology described, as well as uncertainties in the pricing and quantities for labor, materials, and equipment. A contingency covering these and similar factors has been included in the base capital cost estimate. The uncertainty for capital costs ranges from about $\pm 20\%$ to $\pm 45\%$, depending on the facility and equipment, with a median uncertainty of about $\pm 30\%$. The uncertainty in the operating costs for most facilities is estimated to range from $+50\%$ to -25% .

Because of the capital-intensive nature of the nuclear industry, the dollar value of the capital charge uncertainty generally overshadows the dollar value of the operating cost uncertainty for most of the facilities evaluated. A weighted overall uncertainty range was calculated for each unit cost based on the three component uncertainties. A statistical analysis of several example unit cost calculations, assuming a normal random distribution of uncertainty around the three variables, indicates that there is a 95+% probability of being within the total uncertainty range cited for each levelized unit cost.

3.2.8.4 Cost Estimates for Transportation

The unit cost development for waste transport was somewhat different than for other waste management facilities.

Estimates of capital costs of transportation equipment were made assuming the equipment is supplied repetitively by qualified vendors on a competitive basis. The capital cost estimate covers costs for the complete transportation system including the cost of the cask,

(a) Use of the 7% cost of money or discount rate for a Federal project is based on the assumption that a full cost recovery methodology would be adopted similar to that described in DOE/EIS-0015, Vol. 4., where possible charges for AFR storage of spent fuel are described and a 6.5% discount rate is employed. The +3% range encompasses the 10% rate specified in the 1972 OMB circular No. A-94 for use in evaluating government projects. The basis for the private industry and utility discount rates is described in DOE/ET-0028, Vol. 1.

rail car or truck trailer, tiedown system, cooling equipment (if needed), and sun shields. Costs of locomotives and tractors were included in the freight or haulage charges and costs of the waste containers were included in the predisposal waste treatment costs.

The capital costs were translated into unit cask use charges, using the unit cost calculational procedure, private ownership financial parameters and the cask capacity. A cask use factor of 80% (292 days per year) and an annual maintenance charge of 2% of the capital costs were assumed.

Round-trip freight or haulage charges were developed (see DOE/ET-0028, Vol. 4, Section 6) for both rail and truck transportation. A unit freight charge was developed by dividing the freight charge per trip by the cask capacity. The total unit transport cost was obtained by adding the unit cask use charge to the unit freight charge. Additional detail on transportation cost calculations is given in the previously mentioned reference.

3.2.8.5 Research and Development Costs

Costs for research and development have been included in the overall systems costs for waste management developed in Chapter 7.

3.2.9 Physical Protection Safeguard Requirements Assessment

The characteristics of spent fuel, the waste materials and the facilities were reviewed and safeguard requirements were identified for each of the waste management steps considered in this Statement. Results of this assessment are summarized in Section 4.10 for predisposal activities, in Section 5.7 for mined geologic repositories and in Section 6.1 for other disposal alternatives.

Safeguard requirements for plants and materials in the nuclear industry are specified in the Code of Federal Regulations (10 CFR 70 and 10 CFR 73). They include physical protection measures employed to prevent the theft or diversion of special nuclear material, to prevent the willful release of radioactive material, and to prevent the sabotage of nuclear facilities. The principal features of these requirements (10 CFR 73) are the protection forces (guards), physical and procedural access controls, intrusion detection aids, communications systems, and plans for emergencies and strict accountability (10 CFR 70) of all items containing nuclear material including fuel elements and containers of waste. Equipment items, systems, devices, or materials whose failure, destruction or release could directly endanger the public health and safety by exposure to radiation are defined as "vital" (10 CFR 73). Under the existing Code of Federal Regulations, spent fuel and some waste materials in the reprocessing cycle would be classified as vital, and the areas in which they are processed would be vital areas. As such, these areas would require substantial levels of physical protection. For example, Federal regulations specify two independent and successive physical controls over personnel and vehicular entry and exit to and from vital areas.

The required physical protection measures are affected by the potential risk of theft of material that has special strategic worth or is highly radioactive, or by the conse-

quences to the public following sabotage at a facility handling these materials. The level of the potential risk will in turn be determined by the characteristics of these possible targets and the kind and degree of threat anticipated.

Safeguard requirements for the waste management facilities considered in this Statement were characterized based on the attractiveness and accessibility of the wastes as potential targets for theft or sabotage. Attractiveness depends on composition and physical form of the waste. The important aspects of composition are the concentration of fissionable materials and radioactivity. Radioactive wastes are not considered good sources of fissile material for the manufacture of a weapon because of the small quantities of fissile materials per unit volume. Of the waste forms considered in this Statement, only spent fuel contains attractive quantities of such materials. However, the physical condition of spent fuel waste requires sophisticated processing in order to recover the fissile material. Some highly radioactive nuclear wastes may be in a form that would be attractive to an adversary as a source of material that is readily dispersable and, because of the health hazard, could be used to threaten and extort gains from industries or public agencies.

In evaluating the potential for sabotage, consideration was given to design features that could significantly reduce the consequences of sabotage and contribute to the protection of this material. These design features include the thick shielding around the more radioactive process vessels (walls up to 2 m thick); tornado, earthquake and flood protection requirements for all key process facilities; monitored cells and operations; and equipment for detecting and coping with releases of radioactivity. These features generally result in facilities that are unattractive targets for sabotage.

Accessibility of the waste materials was also considered. Factors affecting accessibility include: 1) quantity available at a given location, 2) the degree of isolation of the location, and 3) the complexity of the devices necessary for handling the material (e.g., whether they are operated manually or automatically and whether special knowledge or skills are required).

The final element considered in assessing safeguard requirements was the threat level of potential adversaries. The overall safeguard risk was assessed by considering the above elements--the attractiveness of the material, its accessibility, and the threat level--in the following relationship:

$$\text{Risk to Society} = \text{Frequency} \times \text{Success Rate} \times \text{Consequences}$$

The frequency of attempts, related in part to the attractiveness of material; the success rate, related in part to the availability of the material; and the consequences, measured by effects on the public and the environment, are also all affected by the skills, motivation, financial backing and intrepidity of potential adversaries. All contribute to the risk to society. The relationship shows that if one or more of these factors is very small, the risk to society is also small.

Frequency and success probabilities are difficult to define. However, safeguards measures normally in place for the vital facilities and vital materials of the fuel cycle are

designed to reduce the frequency and success rate to very small values. The safeguard measures will also significantly reduce the consequences of an adverse action through implementation of safeguard emergency plans by providing effective response to threats and attempted adversary actions, and by providing effective assistance to public agencies in protecting the public from the consequences of these threats and actions.^(a)

(a) See Appendix E of 10 CFR 50 and Appendix C of 10 CFR 73.

REFERENCES FOR SECTION 3.2

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3.3 NATURALLY OCCURRING RADIATION AND STANDARDS FOR EXPOSURE TO MAN-MADE RADIATION

Although public awareness regarding radiation has grown markedly in recent years, many readers may not be aware of all of the kinds and quantities of naturally occurring radiation around them. Because of this and because naturally occurring radiation can often be used as a meaningful perspective for evaluating radiation exposure from other sources, a summary of radiation from naturally occurring sources is provided.

To protect workers and the public from excessive exposure to man-made radiation sources and yet realize the benefit from the use of these radiation sources, standards or limits of exposure for various circumstances have been established by several authoritative bodies. Exposures up to these standards are believed not to result in undue risk to the individual. Regardless, the practice of keeping exposures as low as reasonably achievable is fundamental in the radiation protection field. As a consequence, in many facilities the average exposure is not more than one-tenth of the occupational standard. Because of the importance of standards in the control of radiation exposure, a summary of presently applicable standards is also presented.

3.3.1 Natural Radioactivity and Radiation Dose^(a)

Depending on their activities and location, people are exposed in varying degrees to several sources of ionizing radiation found in nature. Cosmic radiation entering the earth's atmosphere and crust is one natural source of exposure. Also, nuclear interactions of cosmic rays with matter produce radiation and radionuclides to which people are exposed. Other sources exposing people to radiation are naturally occurring radioelements in the earth's crust.

Natural radioactivity includes all ionizing radiations and radionuclides except those that have been produced by man's activities, such as that produced by nuclear weapons, bombardment of targets by ion accelerator beams, in nuclear reactors, and from medical and dental x-rays. Sometimes a distinction is made between natural radioactivity in an unmined uranium ore body and "enhanced radioactivity" in mine or mill tailings, for example, radioactivity left on the earth's surface.

The following discussion of dose^(b) and dose rate to the U.S. population from natural radioactivity is presented as perspective for dose estimates associated with management of commercial radioactive wastes in the LWR fuel cycles. No contention is made that exposure to natural radioactivity is or is not harmful. However, when doses associated with waste management are small fractions of natural background dose, such doses would probably be viewed as insignificant.

(a) The discussion of natural radioactivity was taken largely from Natural Background Radiation in the United States, NCRP Report No. 45, Washington, DC, 1975.

(b) Throughout this Statement, the term "dose" may generally be taken to mean the more rigorous term "dose-equivalent." The latter, expressed in units of rem or millirem (one one-thousandth of a rem), implies a consistent basis for estimates of consequential health risk, regardless of rate, quantity, source, or quality of the radiation exposure. Unless otherwise specified, dose is that for the whole body.

3.3.1.1 Cosmic Radiation

Cosmic radiation refers both to primary energetic particles of extraterrestrial origin that strike the earth's atmosphere and to secondary particles generated by the interaction of primary particles with the atmosphere (radionuclides produced by cosmic radiation are discussed later). The primary cosmic radiation consists of particles produced outside the solar system and particles emitted by the sun. The cosmic ray dose rate to the population living at sea level is about 26 mrem per year, taking into account shielding from structures. Considering the altitude distribution of the U.S. population, the average dose rate is 28 mrem per year. In Denver, which is the largest city at a relatively high altitude (1600 meters) in the United States, the average dose rate from cosmic rays is about 50 mrem per year. In Leadville, Colorado (3200 meters), which has a population of about 10,000, the average cosmic ray dose rate amounts to 125 mrem per year. High altitude airplane flights add a small fraction to the population dose from cosmic rays at ground level. For example, a jet flight of 5 hours duration (e.g., transcontinental or transatlantic at 12 km altitude) at mid-latitudes would result in a dose of approximately 2.5 mrem to the whole body. An extreme case would be a 10-hr polar route flight from, for example, California to Europe where the long flight time and the higher cosmic ray intensities at high latitudes would result in a passenger dose of approximately 10 mrem (or 20 mrem for a round trip).

3.3.1.2 Terrestrial Radioactivity

Terrestrial radioactive material is present in the environment because naturally radioactive isotopes are constituents of a number of elements in the earth's crust. The nuclear interaction of cosmic rays with constituents of the atmosphere, soil, and water also produce a number of different radionuclides. These naturally occurring radionuclides give rise to both external and internal irradiation of man.

Cosmogenic Radionuclides

Cosmogenic radionuclides are produced through interaction of cosmic rays with atoms in the atmosphere and in the outermost layer of the earth's crust. The entire geosphere contains radionuclides produced in this fashion. The four cosmogenic radionuclides that contribute measurable dose to man are hydrogen-3 (tritium) (^3H), beryllium-7 (^7Be), carbon-14 (^{14}C), and sodium-22 (^{22}Na), all produced in the atmosphere. The total contribution to the average dose rate (in addition to direct cosmic radiation) by these four nuclides is less than 1 mrem/yr.

Primordial Radionuclides

Several dozen naturally occurring nuclides are radioactive with half-lives of at least the same order of magnitude as the estimated age of the earth (4.5×10^9 yr), and are consequently assumed to represent a primordial inventory (that is, some radionuclides are remaining since the formation of the world). There are three chains or series radionuclides headed by thorium-232 (^{232}Th), uranium-235 (^{235}U), and uranium-238 (^{238}U). These radionuclides decay ultimately to a stable isotope of lead through a chain of decaying nuclides of wide ranging half-lives. These chains contain the, perhaps more familiar,

nuclides radium-226 (^{226}Ra) and radon-222 (^{222}Rn) as well as 31 other radionuclides. Other radionuclides decay directly to stable nuclides. The most significant of the primordial radionuclides in terms of dose is potassium-40 (^{40}K). Aside from a small contribution to dose by rubidium-87 (^{87}Rb), the remainder of the primordial radionuclides, including plutonium-244 (^{244}Pu), occur in extremely small amounts and make no significant contribution to dose. Doses resulting from these primordial radionuclides are discussed below.

External Gamma Radiation. The significant contributors to dose to people from outside of their bodies are ^{40}K and the decay products of the ^{238}U and ^{232}Th series. The principal determinant of outdoor terrestrial radiation at a given location is the soil concentration of natural radionuclides. In addition to soil composition, the radiation outdoors varies depending on the moisture content of the soil, the presence and amount of snow cover, and on the radionuclide concentration in the atmosphere which itself is quite variable. Indoors, the level of radiation is modified by the degree of shielding provided by the building materials against the outdoor radiation, and the amount of radiation originating from radionuclides in the building materials. Variations in outdoor radiation will be partially reflected indoors and, in addition, the contribution from radon decay products will depend on the room air ventilation rate. Each of these factors can play an important role in determining the exposure received by the population.

The overall population-weighted dose rate in the United States from external terrestrial radiation is estimated to be 28 mrem/yr. Moreover, variability in external terrestrial radiation is larger than that for other natural sources of human exposure. This variation in dose rate is characterized by nominal external terrestrial dose rates to the whole body of 15, 30, and 55 mrem/yr for the Atlantic and Gulf Coastal Plains, for the majority of the United States, and for an undetermined area along the Rocky Mountains, respectively.

Internally Deposited Radionuclides. While all natural radionuclides may add to internal (inside the body) radiation doses, only a few are found to be significant contributors. These include ^3H , ^{14}C , ^{40}K , and ^{226}Ra and ^{228}Ra and their decay products. Within the United States, all of these are relatively uniformly distributed so that their levels in foods and water do not vary appreciably with geographic location. In the United States widespread food processing and widespread transportation of foods and people have an additional "averaging" effect on radionuclide contents of diets throughout all geographic areas.

The average total internal whole-body dose rate of about 22 mrem/yr is dominated by about 20 mrem/yr from ^{40}K .^(a) Dose rates to specific organs from internally deposited radionuclides are about 30 mrem/yr to the gonads and other soft tissues, 60 mrem/yr to bone

(a) Potassium is an essential element in the body and is physiologically controlled, hence variations in dietary composition will have little effect on body content or radiation dose received. The same is largely true for the cosmogenic radionuclides ^3H and ^{14}C .

surfaces, and 25 mrem/yr to bone marrow. The dose to women from internally deposited radionuclides is about 25% lower than that to men, because of their smaller potassium content per unit body weight.

Dose to Lung from Inhaled Radionuclides. Dose to the lung from natural airborne radionuclides results principally from the alpha-emitting daughters of ^{222}Rn . The short range of alpha radiation means that the doses are delivered locally to the lung tissue, particularly to the bronchial epithelium. The average dose rate to the total lung is about 90 mrem/yr, while the bronchi epithelium receives about 450 mrem/yr.

Variability in dose rate to the lung is dependent on local concentrations of ^{222}Rn . There is some increase in areas with elevated levels of ^{238}U and ^{226}Ra in soil and a decrease in coastal regions during periods of onshore winds. Levels of ^{222}Rn indoors are dependent on the building's structural materials and ventilation rates. Dose rates to the lungs of smokers from the long-lived decay products lead-210 (^{210}Pb) and ^{210}Po from ^{222}Rn may be up to three times higher than for nonsmokers.

3.3.1.3 Summary of Whole-Body Dose

From the foregoing, the combined whole-body dose rates from terrestrial radioactivity received by groups at 1) sea level for the Atlantic and Gulf Coastal Plains, 2) for the majority of the United States, and 3) for an undetermined area along the Rocky Mountains is 15, 30, and 55 mrem/yr, respectively. The internal and cosmic ray dose rate to the whole body adds about 50 mrem/yr, which results in totals of 65, 80, and 105 mrem/yr as shown in Table 3.3.1.

The whole-body dose rate for groups living at an altitude of 1500 m would be increased by about 20 mrem/yr from the increased cosmic ray radiation. A total whole-body dose rate of 125 mrem/yr from all sources essentially represents the situation for the city of Denver, where both cosmic and terrestrial components are higher than average.

In this Statement, doses calculated as resulting from various waste management activities are often compared with the dose received from naturally occurring sources. To avoid use of ranges of naturally produced doses and to suggest the lack of certainty in the value for any individual, a well-rounded 100 mrem/yr dose rate has been used for illustration. On that basis, the doses used in this report for the population and time periods cited are as given in Table 3.3.2.

TABLE 3.3.1. Summary of Average Whole-Body Dose-Equivalent Rates from Naturally Occurring Radiation, mrem/yr

	Cosmic Rays (Sea Level)	Terrestrial Radiation		Total
		External	Internal	
Atlantic and Gulf Coastal Plains	28	15	22	65
Majority of U.S.	28	30	22	80
Rock Mtn. Area	28	55	22	105

TABLE 3.3.2. Nominal Whole-Body Dose Equivalents from Naturally Occurring Radiation

	<u>Annual Dose</u>	<u>70-Year Accumulated Dose</u>
Individual	0.1 rem	7 rem
Regional Population (2 million)	2×10^5 man-rem ^(a)	1.4×10^7 man-rem
World-Wide Population (6 billion)	6×10^8 man-rem	4×10^{10} man-rem

(a) Man-rem: the sum of the product of the dose received and the number of individuals receiving that dose.

Using the foregoing population doses from naturally occurring radiation and the relationship between population dose and health effects as described in Appendix E (50 to 500 fatal cancers plus 50 to 300 serious genetic defects per million man-rem),^(a) the number of health effects that might be associated with naturally occurring radiation were calculated and are presented in Table 3.3.3.

TABLE 3.3.3. Health Effects Calculated for 70-yr Accumulated Dose from Naturally Occurring Radioactive Sources

	<u>Fatal Cancers</u>	<u>Serious Genetic Defects</u>	<u>Total Health Effects</u>
Regional Population (2 million)	700 to 7,000	700 to 4,000	1,400 to 11,000
World-Wide Population (6 billion)	2,000,000 to 20,000,000	2,000,000 to 10,000,000	4,000,000 to 30,000,000

3.3.2 Applicable Standards for Radiation Exposure Control

A number of existing standards provide for administrative control of potential radiological impacts from waste management operations. These are embodied either in the Code of Federal Regulations (CFR) or comparable codes of state and local governments. Some of these standards are presented here and a more extensive treatment is given in Appendix C.

3.3.2.1 Basic Radiation Standards

The basic radiation standards that apply to all NRC licensees are given in Title 10

(a) Other suggested conversion factors would indicate more effects and others less, not excluding zero effects. The Committee on the Biological Effects of Ionizing Radiation (BEIR), National Academy of Sciences, released in July of 1980 an updated report, the BEIR III report, that indicates risk estimates of cancer death from low levels of radiation are only half what they were thought to be eight years ago (as reported in the BEIR I report, 1972). The range of conversion factors used in this statement encompass the values suggested in both the BEIR I (1972) and BEIR III (1980) reports.

Part 20 of the Code of Federal Regulations (10 CFR 20). Title 10 is based on NCRP, ICRP and FRC guidelines (25 F.R. 4402 et seq May 18, 1960) on radiation standards and the U.S. Government has endorsed the model regulatory code of the United Nations, which closely follows ICRP philosophy. An excerpt from 10 CFR 20 follows:

20.101 Exposure of individuals to radiation in restricted areas.* (a) Except as provided in paragraph (b) of this section, no licensee shall possess, use, or transfer licensed material in such a manner as to cause any individual in a restricted area to receive in any period of one calendar quarter from radioactive material and other sources of radiation in the licensee's possession a dose in excess of the limits specified in the following table:

	<u>rem/calendar quarter</u>	<u>(rem/year)</u>
Whole body; head and trunk, active blood forming organs; lens of eyes, and gonads	1-1/4	(5)
Hands and forearms; feet and ankles	18-3/4	(75)
Skin of whole body	7-1/2	(30)

(b) A licensee may permit an individual in a restricted area to receive a dose to the whole body greater than that permitted under paragraph (a) of this section, provided:

(1) during any calendar quarter the dose to the whole body from radioactive material and other sources of radiation in the licensee's possession shall not exceed 3 rems; and

(2) the dose to the whole body, when added to the accumulated occupational dose to the whole body, shall not exceed 5 (N-18) rems where "N" equals the individual's age in years at his last birthday.

*"Restricted Area" means any area whose access is controlled by the licensee to protect individuals from exposure to radiation and radioactive materials.

Title 10 Part 20 also tabulates limiting concentrations in air and water for many radionuclides, for both the working environment and unrestricted areas, which are not to be exceeded. For individuals in restricted areas, these concentration limits have been calculated, based on continuing exposure for 50 years and standard physiological parameters, to give doses no higher than either those specified above or 15 rem per year to non-specified organs of the body.

For unrestricted areas, standards specify that no individual should receive a dose to the whole body in any one calendar year in excess of 0.5 rem, although some exceptions based on primary concurrent limits (see 10 CFR 20.105) do allow higher doses. In addition, the average dose from all modes of exposure to "a suitable sample of an exposed population group" should not exceed one-third of the limiting dose criteria. Concentration Guides for air and water in unrestricted areas are based on limits of the resultant annual dose to individuals (to either the whole body or specific body organs) of not more than one-tenth the limiting dose for restricted areas.

Since radiation protection guides for the general public are based on averages over a period of 1 year or longer, the evaluation of long-term average exposures should include consideration of reasonable annual occupancy factors as well as the variability of the exposure rates.

3.3.2.2 Other Requirements

EPA Uranium Fuel Cycle Standards

Federal Reorganization Plan No. 3 of 1970 specifically transferred to the Environmental Protection Agency (EPA) the authority to establish standards for "quantities of radioactive materials in the environment." Under this authority, EPA in 1977 issued regulations (40 CFR 190) prescribing "Environmental Radiation Protection Standards for Nuclear Power Operations," which read in part:

190.02 Definitions

(b) "Uranium fuel cycle" means the operations of milling of uranium ore, chemical conversion of uranium fuel, generation of electricity by a light-water-cooled nuclear power plant using uranium fuel, and reprocessing of spent uranium fuel, to the extent that these directly support the production of electrical power for public disposal sites, transportation of any radioactive materials in support of these operations, and the reuse of recovered non-uranium special nuclear and by-product materials from the cycle.

190.10 Standards for Normal Operations

Operations covered by this Subpart shall be conducted in such a manner as to provide reasonable assurance that:

(a) the annual dose equivalent does not exceed 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ of any member of the public as the result of exposures to planned discharges of radioactive materials, radon and its daughters excepted, to the general environment from uranium fuel cycle operations and to radiation from these operations.

(b) the total quantity of radioactive materials entering the general environment from the entire uranium fuel cycle, per gigawatt-year of electrical energy produced by the fuel cycle, contains less than 50,000 curies of krypton-85, 0.5 millicuries of iodine-129, and 0.5 millicuries combined of plutonium-239 and other alpha-emitting transuranic radionuclides with half-lives greater than one year.

By definition these regulations do not apply to transportation or operations at waste disposal sites but do apply to reprocessing of spent uranium fuel for reuse in the generation of electricity.^(a) Where applicable these regulations supersede the related portions of 10 CFR 20. The basis for the numerical values given was a cost/benefit analyses of expected reductions of estimated environmental doses and consequent "health effects" versus estimated dollar costs of additional effluent treatments.

Clean Air Act Amendments of 1977

The 1977 amendments to the Clean Air Act specifically required the EPA Administrator to determine whether emissions of radioactive pollutants will cause or contribute to air pollution which may endanger public health. The Administrator has made an affirmative

(a) EPA is presently developing radiation protection standards for the disposal of high-level waste. In addition, NRC has published an Advanced Notice of Proposed Rulemaking relative to their technical criteria for geologic disposal of high-level waste.

finding and listed radionuclides as hazardous air pollutants under Section 112 of the Act (44 FR 76738, December 27, 1979). EPA must now propose regulations establishing emission standards for radionuclides.

Marine Protection, Research and Sanctuaries Act of 1972. Public Law 92-532

Dumping of any material into ocean waters is permitted only pursuant to a permit from EPA, or, for dredged material, the Corps of Engineers. The Act specifically precludes issuance of a permit for dumping of high-level radioactive waste.

Department of Energy Requirements

Other than the quarterly fractionation of the Nuclear Regulatory Commission dose limits, and with minor exceptions for specific body organs, the limiting dose criteria of 10 CFR 20 are the same for Department of Energy operations, as given in ERDA Manual Chapter 0524 (ERDA 1975). Any new facilities for commercial high-level waste management are expected to be licensed by the Nuclear Regulatory Commission.

State Regulations

Under Section 274 of the Atomic Energy Act of 1954 as amended, a number of states and the Nuclear Regulatory Commission have executed agreements that permit a state to grant licenses for the control of specified nuclear activities within the state boundaries. Production and utilization of special nuclear materials and Federal facilities are specifically excluded. Examples of state-licensed activities are the commercially operated low-level waste burial sites at Barnwell in South Carolina and at Hanford in Washington. Although each agreement state may establish its own inventory limits and administrative, surveillance, and reporting requirements, the same basic radiation protection standards apply as for Federally licensed facilities. Further, under provisions of the Clean Air Act Amendments of 1977, the states may set standards for radioactive emissions in the air which are more stringent than Federal standards.

EPA Waste Management Standards

The Environmental Protection Agency is responsible for developing standards applicable to all Federal radioactive waste management programs; these standards will be implemented in NRC regulations. EPA has published for public review the initial formulations of their standards.

In commenting on the draft of this Statement the EPA stated that they are presently proposing criteria and standards for radioactive waste management. These criteria and standards will be applicable to any disposal of high-level waste or spent nuclear fuel.

NRC Rules for Licensing of Geologic Repositories

The Nuclear Regulatory Commission has the statutory authority to license facilities used primarily for the receipt and storage of high-level radioactive wastes resulting from activities licensed under the Atomic Energy Act of 1954 and the Energy Reorganization Act of 1974. The Commission has indicated that regulations covering the licensing of De-

partment of Energy disposal facilities will be issued as Part 60 of Chapter 10 of the Code of Federal Regulations (10 CFR 60). The procedural part of the NRC regulations was published for comment on December 6, 1979. It is expected that the technical portion of the regulations will be published for comment in late 1980.

DOT Regulations

Regulations governing the packaging, labeling, and shipping of radioactive materials, including radioactive wastes, are given in Title 49 of the Code of Federal Regulations and are too voluminous to be reproduced here. Included are descriptions of approved shipping containers for various quantities and types of radioactive materials, including performance criteria for protection against accidental damage. Limits on external levels of radiation are provided.

REFERENCES FOR SECTION 3.3

Title 10 Code of Federal Regulations. Part 20.

Title 40 Code of Federal Regulations. Part 120.

Federal Register. Vol. 25, p. 4402 et seq, May 18, 1960.

U.S. Energy Research and Development Administration. 1975. "Standards for Radiation Protection." In ERDA Manual of Operations, Chapter 0524. U.S. ERDA, Washington, D.C.

3.4 RISK AND RISK PERSPECTIVES

The potential environmental impact of nuclear waste isolation is often judged on the basis of a variety of risk and/or perceived risk issues. In this Statement, risk is defined as "probable loss." It is defined as the sum product of the magnitude of losses (the consequences) and the probability that these losses will occur. As defined, it does not discriminate between present or future events or between those of low probability/high magnitude and of high probability/lesser magnitude. Ordinary use of the term risk is not always consistent with this definition. For example, events of large magnitude, no matter how improbable, may be termed a large risk simply because of the size of the consequence. Similarly, when considerable uncertainty surrounds the estimate of probability or consequence, it might be said that a large risk is present. In both of these cases, the expected or most probable loss may be quite low.

Historically, society has tended to concentrate on minimizing the occurrence of high consequence events while giving little attention to low consequence events. An example is the required FAA safety certification of airplanes versus the relatively minor safety requirements for automobiles (seatbelts, safety glass, etc.). Americans are killed by the tens of thousands per year in auto accidents and by hundreds in airplanes. Yet it appears much more attention if not concern is given to 100 plane deaths than to 100 auto deaths. There is justification for placing attention on potential catastrophic events if such events could affect society's ability to recover from the catastrophic events. However, it is important to keep in mind that the amount of risk is not the only consideration in society's assessment of risk. Consideration of the benefit associated with that risk (or why the risk is being taken) also places the risk in perspective. The risk analyses in this Statement do not attempt to quantify the benefit associated with the generation of electricity which results in the production of nuclear waste.

This Statement considers the societal risk of the predisposal waste management technologies, the risk of operating a repository and the risk of long-term loss of containment or isolation. Two approaches to analyzing long-term risk are presented below: comparative hazard indices for both radioactive and non-radioactive materials including nuclear wastes, and the long-term analysis and risks associated with various scenarios for the release of radionuclides from deep geologic burial to the biosphere (consequence studies).

3.4.1 Hazard Indices

Hazard indices are based on estimates of potential risk of released radionuclides compared to other risks. The hazard indices can show whether the quantities of toxic radioactive waste exceed the toxic quantities of other chemicals and substances routinely handled in our society. A number of hazard indices have been developed which are useful in varying degrees in characterizing the risk. They are summarized in Appendix H of Volume 2. Hazard indices associated with radioactive materials are considered useful to the extent that the comparisons inform the reader about the magnitude of hazard compared to more familiar hazards.

One such hazard index is based on the amount of water required to bring the concentration of a substance to allowable drinking water standards. In the present case the amount of water required to bring the quantity of uranium ore (0.2% U_3O_8) necessary to make 1 MT of reactor fuel to drinking water standards (7×10^{-2} g/l) was used as a basic hazard index. Assuming enrichment of ^{235}U to 3%, about 3,400 MT of ore would be required (95% recovery to make 1 MT of fuel. The hazard index of natural uranium of this quantity of ore is 8.7×10^7 m³. The hazard index of the radionuclides in 1 MT of spent fuel was calculated based on 10 CFR 20 drinking water standards and summed for various times after the spent fuel was removed from the reactor. The hazard index for high-level waste from uranium-plutonium recycle was calculated in a similar way. Division by 8.7×10^7 m³ made the hazard index relative to 0.2% uranium ore. In addition the hazard index of various ores was calculated relative to the volume of uranium ore equivalent to 1 MT of reactor fuel. These indices are presented in Table 3.4.1.

TABLE 3.4.1. The Relative Toxicity (Hazard) of Various Ores Compared to U Ore (0.2%)

<u>Type of Ore</u>	<u>Average Ore</u>	<u>Rich Ore</u>
Arsenic	1	10
Barium	5	20
Cadmium	28	120
Chromium	170	230
Lead	40	100
Mercury	460	3800
Silver	1	7
Selenium	70	220

The hazard index for spent fuel and high-level waste is shown in Figure 3.4.1, together with similarly developed hazard indices for ranges of common ores.

As seen in Figure 3.4.1 the hazard index for spent fuel or reprocessing waste from uranium-plutonium recycle relative to the ingestion toxicity of the volume of 0.2% uranium ore necessary to produce 1 MT of reactor fuel is on the order of that for rich mercury ores at about 1 year after removal of the spent fuel. The hazard index is on the order of that for average mercury ore at about 80 years. By 200 years the index is about the same as average lead ore. By 1500 years the relative hazard index for high-level waste is the same as the ore from which the fuel was made. For spent fuel the relative hazard index is about the same as the ore from which it came at about 10,000 years.

It is not suggested that spent fuel or high-level waste are not toxic. They are highly dangerous if carelessly introduced into the biosphere. It is, however, suggested that where concern for the toxicity of ore bodies is not great, then spent fuel or high-level waste should cause no greater concern particularly if placed within multiple-engineered barriers in geologic formations at least as, if not more, remote from the biosphere than these common ores.

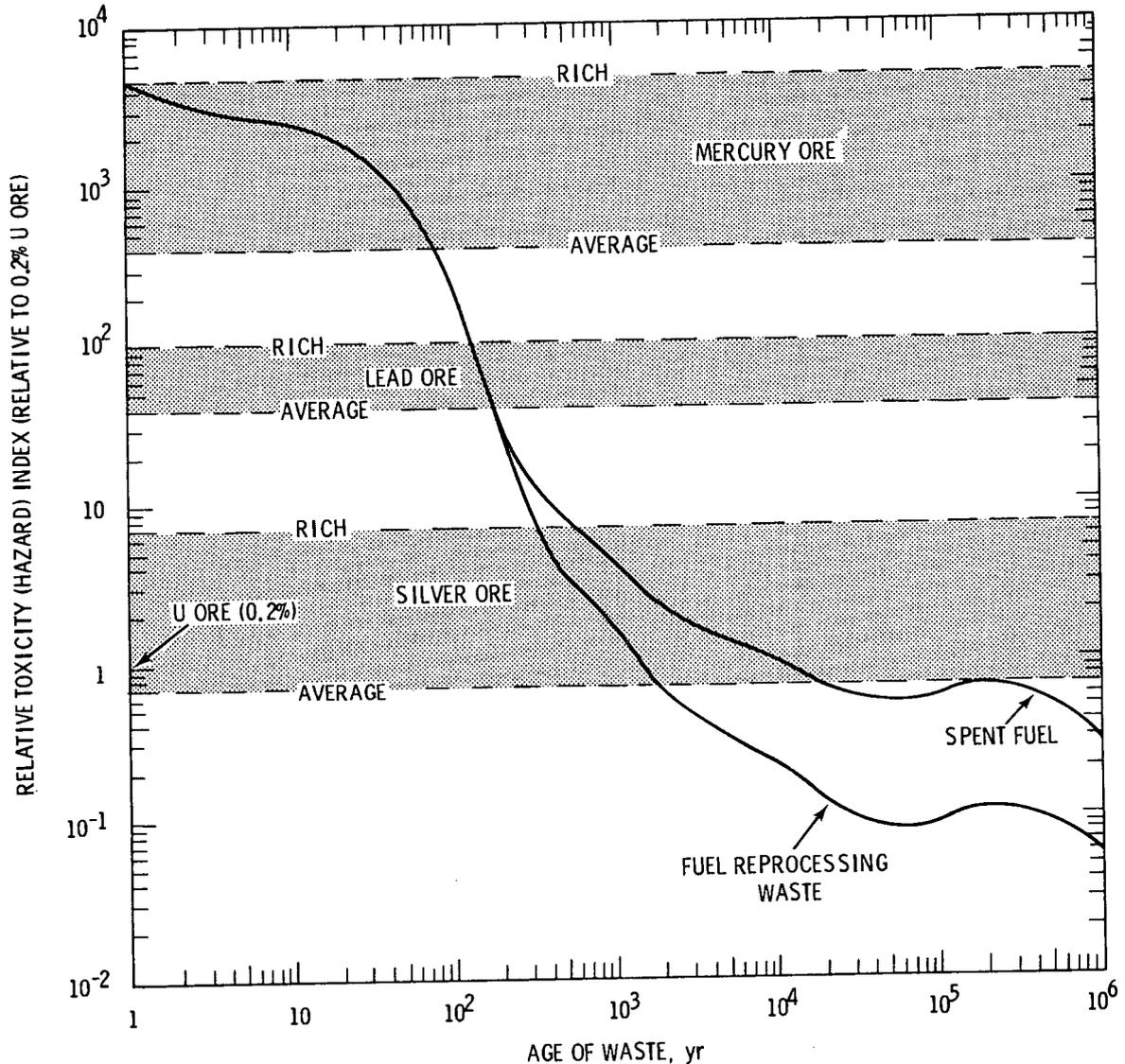


FIGURE 3.4.1 Toxicity of Spent Fuel and Reprocessing Waste from Uranium-Plutonium Recycle Relative to 0.2% Uranium Ore Necessary to Produce 1 MT of Reactor Fuel

Hazard indices generally neglect major confinement features such as the waste concentration (Hill 1977, Lash 1976), release mechanisms and dynamics (de Marsily 1977), and aspects of the food chain pathways. The hazard indices for the most part do not characterize the population exposures associated with conceivable natural and man-induced disruptive events--the key aspects of a risk assessment.

3.4.2 Consequence Analysis and Risk Assessment

Consequence analysis is the estimation of the effects of postulated accidental releases of radionuclides. Risk assessment is the calculation of the consequences of the spectra of possible accidental releases multiplied by their probabilities and summed to give a total risk. In this sense, the EIS does not present a complete risk assessment. The technique for such an assessment is still under development.

Since long-term repository containment cannot be demonstrated by short-term test, mathematical models must be relied on to predict the long-term behavior of the repository. Risk assessment is thus dependent on the development of reasonable predictions of the long-term behavior of the processes and phenomena that could occur within the repository system. The risk assessment under development for geologic isolation is taking the form described in the following methods.

3.4.2.1 Disruptive Events

Many geologic events and processes occur because of the long-term motion of the earth's plates with their associated stresses and strains, and by the action of long-term weather patterns associated with a variety of astrophysical and earth phenomena. Many of these phenomena are predictable (usually with an element of randomness); others can only be assigned an estimated site-dependent probability of occurrence. More specifically the key interest in predictive modeling is whether a site (selected by virtue of historical stability) will change to an unstable area (e.g., active faulting, volcanism, significant ground- and/or surface-water activity, etc.).

Potential disruptive phenomena that could affect a repository have been categorized as natural processes, natural events, man-caused events and repository-caused processes and are listed in Table 3.4.2.

The science of geology has tended to concentrate on predicting the location of ores and fossil fuels and to explain the structure of the earth. Nuclear waste isolation appears to be the first subject of large interest in long-term predictive geology. Many geologists have recently been engaged in the development of suitable predictive geologic models and/or scenarios. This research is concentrating on specific sites as well as global processes.

To be complete, risk assessment must include all significant sources of risk and must predict the condition of the repository and surrounding area following failure, the time of failure occurrence and its probability of occurrence. This evaluation is called "Scenario Analysis" (Burkholder 1978, Greenborg et al. 1978). In general, these evaluations employ models that are very complex and require the capabilities of electronic data processing. Confidence in the models can be increased by comparing the results of the models to natural systems which exist and adjusting the models until a reasonable degree of conformance is reached. This concept of calibration and verification has been employed in the hydrology models discussed below.

3.4.2.2 Lithosphere/Atmosphere Transport

This risk assessment process includes both lithospheric (by ground water) and atmospheric (by airborne and other surface processes) radionuclide transport analysis. The physicochemical processes governing ground-water movement and transport of pollutants are sufficiently understood that mathematical models can be formulated. However, these models require measured physicochemical parameters representing the specific site in order to simulate the system. These data are seldom adequate in terms of quantity and quality. However,

TABLE 3.4.2. Potential Disruptive Phenomena for Waste Isolation Repositories

<u>Natural Processes</u>	<u>Natural Events</u>	<u>Man-Caused Events</u>	<u>Repository-Caused Processes</u>
<ul style="list-style-type: none"> ● Climatic Fluctuations ● Sea Level Fluctuations ● Glaciation ● River Erosion ● Sedimentation ● Tectonic Forces ● Volcanic Extrusion ● Igneous Intrusion ● Diapirism ● Diagenesis ● New or Undetected Fault Rupture ● Hydraulic Fracturing ● Dissolution ● Aquifer Flux Variation 	<ul style="list-style-type: none"> ● Flood Erosion ● Seismically Induced Shaft Seal Failure ● Meteorite 	<p>Improper Design/Operation:</p> <ul style="list-style-type: none"> ● Shaft Seal Failure ● Improper Waste Emplacement <p>Undetected Past Intrusion:</p> <ul style="list-style-type: none"> ● Undiscovered Boreholes or Mine Shafts <p>Inadvertent Future Intrusion:</p> <ul style="list-style-type: none"> ● Archeological Exhumation ● Weapons Testing ● Nonnuclear Waste Disposal ● Resource Mining (mineral, hydrocarbon, geothermal, salt) ● Storage of Hydrocarbons or Compressed Air <p>Intentional Intrusion:</p> <ul style="list-style-type: none"> ● War ● Sabotage ● Waste Recovery <p>Perturbation of Ground-water System</p> <ul style="list-style-type: none"> ● Irrigation ● Reservoirs ● Intentional Artificial Recharge ● Establishment of Population Center 	<p>Thermal, Chemical Potential, Radiation, and Mechanical Force Gradients:</p> <ul style="list-style-type: none"> ● Induced Local Fracturing ● Chemical or Physical Changes in Local Geology ● Induced Ground-water Movement ● Waste Container Movement ● Increase in Internal Pressure ● Shaft Seal Failure

those data that can reasonably be obtained can be combined with a model to gain valuable insight. Some ground-water and transport models have been calibrated (Gupta and Pinder 1978, Kipp et al. 1976, Cole 1979) through adjustments of parameters to simulate measured behavior and thus can be used with some confidence in forecasting. These models have also been verified (Kipp et al. 1976, Ahlstrom 1977, Robertson 1977) by showing that they duplicate past trends in water table changes and contaminant transport in field situations.

Similarly airborne transport of ejected or reentrained radionuclide aerosols, subsequent uptake by biota, food chain pathways and exposure to and ingestion by man can be evaluated for specific sites.

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3.5 NONTECHNICAL ISSUES

Many of the issues concerning the management and disposal of radioactive waste do not confine themselves to strictly technical aspects of the problem. "Nontechnical issues" refers to broad social, political, and institutional concerns. This discussion is, in large part, based upon a Conference on Public Policy Issues in Nuclear Waste Management and on a recent report (Hebert et al. 1978).

The first part of this discussion organizes the nuclear waste issues into a smaller subset of issues and describes various positions on the issues. Further, the response to the issues raised by government agencies is discussed. The second part of this discussion examines in detail two areas of concern: short-term institutional arrangements and institutional arrangements for the long term.

3.5.1 Social Issues

A major issue concerning some people is the balancing of risks and benefits between this generation and future generations. One position on the issue is: at present transformation of the long-lived radioactive wastes into more short-lived forms is not feasible. As a result, future generations will have a burden of surveillance and monitoring, of risk to health and safety, and of corrective action should a containment breach occur, either from human or natural causes. Those holding this view state since this burden is difficult to specify and since the nation can afford to forego nuclear power benefits, production of more wastes would be morally irresponsible. An opposite position stresses that the risk exported to future generations is not unique to radioactive waste, is lower than commonly accepted risks, is a threat to relatively few people, and is low because of manmade and geologic barriers. Such low risk does not constitute an unfair burden given the benefits of nuclear power. A third position on this issue takes a more global view. Those with this view state that the issue of waste should be considered in the context of the benefits and costs and risks of all energy sources, not just nuclear power. For example, the problem of nuclear wastes should be viewed in the context of the benefit of preserving fossil fuels for future generations.

The issue of distribution of risk between generations is being examined by the Department of Energy and also by EPA and NRC. Early draft criteria by EPA have been explicitly concerned with this problem and reviewed in a public workshop held in Denver on March 30, 1980 (43 FR 2223). In his February 12, 1980 message on waste management, the President stated that his paramount objective is to "protect the health and safety of all Americans, both now and in the future." The Department of Energy in its Statement of Position on the Waste Confidence Rulemaking Hearings (DOE-NE-0007) takes recognition of this issue in its stated performance objectives, especially Objective 2, which specifies isolation for 10,000 years with no prediction of significant decrease in isolation thereafter, and Objective 5, which stresses conservatism in technical approach to provide assurance that regulatory standards can be met.

A second issue involves the need for candor. Concern has been expressed that information provided by the government and the nuclear industry concerning such events as the leaks at the Hanford, Washington, site has not been timely or relevant. However, since the mid-1950s there has been a large number of technical articles on nuclear power. Some take this as evidence of candor, while others see the flood of articles as an attempt to confuse the layman and increase reliance on the technical expert.

The President, in his February 12, 1980 message, noted that past governmental efforts to manage radioactive wastes have neither been technically adequate, nor have they sufficiently involved states, local governments and the public in policy and program decisions. The message established a program with mechanisms for full participation of these groups and continuous public review. The Department of Energy is fully committed to this program.

A third issue, public involvement, was a major topic at the Conference on Public Policy Issues (NSF 1976). Panelists at this conference generally agreed with the position that any person, group, or institution wanting to be involved in nuclear waste policy decisions has that right. Conference participants also pointed out that public participation does not guarantee sensible decisions nor an enhanced understanding of the issue. While general agreement was that final decisions should rest with the Federal government, some urged very strong public input on nuclear waste decisions via such mechanisms as state initiatives.

As stated above, the President's message has mandated full public participation in waste management policy decisions. Prior to this message, the Department of Energy held five public meetings in various regions of the country to seek public comment on the draft of this Statement in addition to the usual written comments. As a result of this input, this Final Statement has undergone extensive revision. Volume 3 of this Statement documents the extent of this revision. Further, the Interagency Review Group (IRG) received extensive public comment on their report dealing with nuclear waste management policy.

A fourth issue is that of uncertainty. Uncertainty pervades the technical and non-technical discussion about nuclear waste. The major uncertainties relating to nuclear waste involve: 1) effects of small doses of radiation received at low dose rates over a long time, 2) uncertainty about the ability to isolate nuclear wastes from the biosphere, and 3) uncertainty about human fallibility and malevolence. Some react to the uncertainty with caution and may urge a go-slow approach to waste isolation, while others feel that the uncertainties are sufficiently low to proceed with a waste isolation and disposal program.

In its Statement of Position for the "Waste Confidence" Rulemaking (DOE-NE-0007) the Department of Energy proposes a technically conservative approach to compensate for the perceived uncertainties in the ability to predict natural phenomena over long periods of time. The approach will utilize conservative design parameters, large margins for error, and multiple engineered and natural barriers in a step-by-step approach to implementation which will permit the capability of corrective action, should processes not operate as expected.

A fifth issue is that of equity. Some feel that those who live near a waste repository may be said to bear a greater risk in proportion to their benefit than do those remote from

the repository. Some feel that those near the repository may not even benefit from the nuclear power which produced the waste. Another position stresses that people indirectly benefit from nuclear power because they buy products made with electricity from nuclear power and, therefore, such equity issues are less valid.

The Department of Energy is considering the feasibility of regional repositories, (i.e., repositories which serve the needs of the surrounding region) partly in response to concerns about equity (see discussion in Section 5.3). Under various scenarios there will be a need for more than one repository for a nuclear economy of 250 GWe by year 2000 (e.g., Case 3 in Section 3.2).

Concern about safeguards is a sixth issue. This concern hinges largely, though not exclusively, on the fact that plutonium, produced in the process of nuclear power production, is used in nuclear weaponry. Commercial fuel cycles which separate plutonium or other material with potential use in weapons raise the concern that they might be used for clandestine weapons development. Accounting for such material has been seen by some as inadequate. Some also worry that security against nuclear threats can only be achieved by intolerable infringements on personal freedom, while others feel that this is not the case. There is also a large difference in the perception of how difficult it is to build a bomb, ranging from the belief that one only needs access to a public library to a belief that it is a highly risky and technically challenging task requiring a sophisticated manufacturing capability.

The Department of Energy has an active research program for developing and improving safeguard and physical security methods that deal with transportation, storage and handling of radioactive materials. The NRC has promulgated and enforced safeguards and physical protection regulations for special nuclear materials such as plutonium (10 CFR 73).

Alternatives to nuclear power form a seventh issue area; that is, how one perceives conservation and other energy production alternatives affects perceptions of nuclear waste. The belief that cheaper, safer, less-polluting alternatives to nuclear power are available would incline the holder of that belief to oppose the production of nuclear wastes. Some, however, feel that nuclear power is superior to currently available technologies and therefore are willing to accept the radioactive waste problem. Even if no further nuclear weapons production or power generation occurred, an inventory of wastes from past activities would need to be stored or disposed.

In its Statement of Position at the "Waste Confidence" Rulemaking, the Department of Energy proposed in Objective 7 that disposal concepts selected for implementation should be independent of the size of the nuclear industry (DOE/NE-0007). This is in accord with the President's statement of February 12, 1980, which requires that waste disposal efforts proceed regardless of future developments in the nuclear industry. This EIS examines 5 cases of nuclear development ranging from termination of nuclear power in 1980 to full development to properly assess nuclear waste management systems (see Chapter 7).

An eighth issue area is the transportation of nuclear waste material. Concerns about accidents, sabotage, and thefts of material in transit are at the core of these concerns and so relate to the issues previously mentioned.

The U.S. Department of Transportation (DOT) is currently in a rulemaking process concerning transportation of high-level nuclear wastes (45 FR 7140). Further, current regulations of both DOT and NRC are considered to adequately protect public health and safety (49 CFR, Parts 173 and 177).

The irreversibility of geologic waste disposal is the core of a ninth issue. The argument has been made that because of its apparent irreversibility we should delay implementing geologic isolation until we are more certain that the wastes will not be used now or in the future. Other arguments for delay include keeping the wastes retrievable for 20 to 30 years in case something goes wrong in the repository or in case a better method is devised in this period. However, the argument has also been made that disposal methods that are technically impossible to reverse offer the best solution to isolating the wastes from man.

In its Statement of Position (DOE-NE-0007) of April 1980, the Department outlined its "step-wise" approach. This conservative approach would store a limited quantity of material under well understood conditions and then proceed in a series of small steps so that the material could be retrieved should unanticipated problems make the system unacceptable. NRC has also reflected this approach in a recently issued draft of possible technical regulations which would require the capability of retrievability for 50 years after emplacement operations have ceased. The ability to retrieve the wastes during the initial periods of operation is seen as one of the main advantages of mined geologic repositories.

The tenth issue area involves the distinction drawn between commercial and military wastes. Some have argued that no distinction should be made on the constraints of the management of the two wastes, while others have argued that they should be kept distinct because of the very different physical nature of the wastes.

The Presidential message of February 12, 1980 specifically directs that the radioactive waste management program seek to isolate and dispose of wastes from both civilian and military activities.

International responsibilities form an eleventh area of concern. The waste issue is larger than U.S. boundaries because of technology export and import of wastes and because of possible international solutions to the waste problem. Worldwide releases of radioactivity may cause health and genetic problems which respect no national boundaries. Further, concern has been expressed that in lesser developed countries, cost concerns could lead to an inadequate waste management plan. Since much of the nuclear waste is now produced in foreign reactors, some of which are U.S. exports, the argument has been made that the U.S. must show leadership in solving the nuclear waste problem. An international waste management authority has been proposed to handle these problems.

The Department of Energy is mindful of international responsibilities for nuclear waste and is participating in a number of bilateral and multilateral programs to deal with nuclear waste. Examples are a cooperative investigation with Sweden at a mine in Stripa, Sweden, a

cooperative agreement with the Federal Republic of Germany for exchange of technical information on waste disposal, and active participation in the International Atomic Energy Agency (IAEA).

A twelfth issue area is that of cost of waste management. Participants in the Conference on Public Policy Issues on Nuclear Waste Management showed general agreement that we must be willing to pay for an adequate disposal system. Some fear that adequate charges will not be assessed to provide perpetual care. Current regulations require a fee to be paid to the government at the time of transfer of the waste to Federal custody, although the size of this fee has not been determined.

The President's message of February 12, 1980 specified that "all cost of storage, including cost of locating, constructing and operating permanent geologic repositories will be recovered through fees paid by utilities and other users of the services and will ultimately be borne by those who benefit from the activities generating the wastes."

A final issue area, discussed more fully below, concerns institutions for controlling and managing nuclear waste. These concerns relate both to the short term, i.e., the period of time up to the closure of a waste repository, and to the long term, i.e., the period following closure for the hundreds of years during which the potential hazards of the waste remain. Some individuals contend that past mishaps and leaks involving military wastes are a basis for regarding the current institutional arrangements as inadequate. Others judge that current institutions have done an adequate job or that new arrangements will lead to better waste handling. Further the ability of institutions to monitor disposed waste in the long term is a key part of the issue area. Some feel that technical considerations will make such long-term monitoring unnecessary, while others feel that the waste has to be monitored for as long as 200,000 years and would be a formidable task. A more intermediate view is that monitoring might be required for several hundred years.

In the Department of Energy's Statement of Position for the NRC "Waste Confidence" Rulemaking (DOE/NE-0007), a proposed objective of the program was to provide reasonable assurance that wastes will be isolated from the environment for at least 10,000 years with no prediction of significant decrease in isolation beyond that time. Further governmental concern for this issue is shown by the proposed EPA criterion that a waste disposal system cannot rely on human institutions for a period of more than 100 years (42 FR 53262).

3.5.2 Institutional Issues

The following two sections briefly expand on short-term and long-term institutional concerns. These two sections discuss institutional concerns without reference to scale of the waste management system. Some have argued that institutional issues may potentially become much more severe with increasing scale (LaPorte 1978).

3.5.2.1 Short-Term Concerns and Institutional Design

Technical solutions to waste management problems are not self-implementing. They require institutions, either those existing or ones yet to be created, to make them work.

Setting up a waste management program therefore requires institutional choices: whether to rely on existing organizational arrangements or to develop new ones. This section discusses some considerations regarding choice of one or another set of organizational arrangements for waste management. Additionally, the institutions discussed below should function in conjunction with the engineered design as part of the overall waste management system.

The Department of Energy (DOE) is currently responsible for establishing programs leading toward the treatment, storage, and disposal of nuclear wastes. The Environmental Protection Agency is responsible for setting generally applicable environmental standards for radioactive waste (3 CFR). The Nuclear Regulatory Commission is responsible for implementing these standards, establishing regulations and policies, and licensing commercial waste management facilities (10 CFR 20 301, 42 U.S.C. 5842). State governments (in agreement states) license and regulate low-level burial sites (42 U.S.C. 2021). The Department of Transportation (DOT) shares responsibility for regulation of the transportation of wastes with NRC (38 F.R. 8466, March 22, 1973).

A number of organizational options are available for the management and disposal of nuclear waste. Below are listed four such options: 1) Federal agency; 2) government corporation; 3) government-owned, contractor-operated facility; and 4) contractor-owned, contractor-operated facility. In a Federal agency, waste management functions would be performed directly by Federal agency employees who are ordinarily members of the Federal civil service. A government corporation is a Federally chartered organization with its own legal personality distinct from that of the Federal government. It is exempt from civil service rules, thus allowing the managers of the corporation to retain control over all aspects of personnel management. A government-owned, contractor-operated arrangement is similar to the government corporation, especially in the private contractor's flexibility with respect to personnel practices and financial systems. A contractor-owned, contractor-operated arrangement differs chiefly in that the contractor's financial commitment is much heavier than under a government-owned, contractor-operated arrangement.

In addition to consideration of organizational options, a knowledge of the basic regulatory functions is useful in assessing the adequacy of institutional arrangements for managing and disposal of nuclear waste. The function of regulating the commercial nuclear waste management system includes the tasks of standard-setting, licensing, technical review, inspection, and enforcement. Below is a brief discussion of each task.

Standard-setting and licensing are often done by the same organization. Sometimes, however, one agency (such as EPA) has the task of setting general rules for how tasks must be done (performance standards), while another agency (such as NRC) has the task of applying those general standards to a specific case, and of granting a license to operate when proper conditions have been met.

A technical review of a proposed action for its scientific adequacy may increase the safety of the waste management system by helping to avoid errors at key decision points. Reviewer independence is a valuable attribute; it reduces the opportunities for bias and, hence, the chances that a review will become automatic approval.

Inspection, the regular checking of the actual waste management operation to ensure that it is being performed in the proper manner, is one of the most critical functions in the entire waste management system. If other parts of the system break down, a good inspection system will detect them. If the inspection system itself fails, no one will know whether or not the waste management system is reliable.

The character of the enforcement function depends on whether private or public organizations are the target. In the case of private organizations, credible penalties, such as fines and license revocation, are available. But these sanctions cannot be expected to have the same effect on public organizations, which are less influenced by economic incentives.

3.5.2.2 Institutions in Long-Term Nuclear Waste Management

A number of concerns have been raised regarding the role that human institutions may have in the long-term management of nuclear wastes. Controversy exists concerning: 1) the need for any human institutions to be involved in long-term management, and 2) whether human institutions could actually carry out any functions that might be required of them over the long term.

These discussions are speculative. Historical examples of the behavior and durability of human institutions are the only data that can be applied to the speculations about the potential future stability and performance of institutions. However, to predict what the world will be like 50 to 100 years from now, let alone in several centuries, is very difficult.

Human institutions might enhance safety by accurately predicting the occurrence of the natural events which could compromise the repository (e.g., earthquakes, floods), and in responding to them to reduce consequences. Control over these massive events is not likely.

Human actions that might produce a release of radioactive material from a repository have been grouped into three categories: 1) major catastrophic events, such as nuclear war, 2) direct action against the repository, such as sabotage, drilling and exploration, and excavation, and 3) lapses in monitoring, such as being unaware of a breach in the containment.

Three sets of factors appear pertinent in assessing the institutional role in long-term waste management: 1) the functions that can or should be performed by the institutions, 2) the subjective need for these functions, and 3) the likelihood that the functions will be performed at any given point in time.

Three general categories of functions might increase the safety of a waste repository and mitigate the consequences of potential accidents:

1. Control and management--including monitoring of security and physical integrity, performance of routine physical plant maintenance, and maintenance of a staff of people qualified to carry out technical tasks at the disposal site.

2. Monitoring--including observation of seismic, thermal, and radiological conditions to detect any releases or significant changes in site integrity.
3. Information transfer--including maintenance of records and data about the repository and its contents. Such information would be needed to effect repair of a site, to warn future generations about the dangers of the wastes, to inform people about the resource value of the contents, and to prevent an intrusion into the repository at some time in the distant future.

It has been suggested that human institutions could provide an increment of safety if monitoring, surveillance, and security operations are carried out during the first few centuries after a repository is closed. Human activities would provide a backup to the engineered system. This backup system would have the function of predicting the occurrence of natural hazards, preventing human intrusions, and responding to any anomalies that occurred at repository sites. These last two functions were seen by some to be especially significant in the mitigation of repository accidents.

Predictions are very difficult to make with certainty about whether future societies would find the task worthwhile to support institutions to carry out the functions noted above. It has been argued that it is up to future generations to decide for themselves whether to carry out these functions. Predictions are also impossible to make on whether information can be conveyed across millenia, or whether organizations can be established that could last for such time periods. The focus of assessment has been to analyze any evidence to suggest that if organizational and institutional continuity were necessary, could institutions be established in the present that might survive long enough to carry out their tasks?

The analysis of these issues is, of necessity, purely speculative, and based on historical examples that provide no firm basis for making predictions. However, some examples suggest that complex information in abstract form can be maintained over thousands of years.^(a) The sacred books of major religions and the hieroglyphics of ancient Egypt are examples. Furthermore, many functional organizations, such as the U.S. Government, have survived for a century or more while carrying out roughly the same tasks. A few, such as the British political system, have survived for nearly a millenium. Of course, how much information has been lost in historical times is not known.

The principal conclusions of this analysis are:

- There are no reasons in principle to indicate that human institutional functions cannot survive for hundreds of years, given reasonably stable political systems. However, no strong evidence exists that such functions will, in fact, survive.
- Technical information can be maintained for a very long time if a culture remains literate and the information has a continuing utilitarian value.
- Waste management systems adopted in the present time period should place minimal, if any, reliance on any human management after the repository is closed.

(a) Additionally, no prior known civilization has had both the mass education and communication systems that presently exist.

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CHAPTER 4

PREDISPOSAL SYSTEMS

After radioactive wastes are generated and before their disposal, several predisposal operations are required. The combination of these operations is referred to in this Statement as the predisposal system. The system operations include treatment and packaging to prepare the waste for the specific requirements of a disposal option, interim storage if the treated waste cannot be shipped immediately to a disposal site, and shipment to interim storage and/or to a disposal site. Decommissioning of the waste management facilities, although not a predisposal operation, is discussed in this chapter because it produces wastes which must be managed in a manner similar to those wastes produced by fuel reprocessing and MOX fuel fabrication plants.

This chapter provides examples of processes and facilities that could be used to carry out these predisposal operations for both the once-through cycle and the reprocessing cycle. The processes and facilities described here are not dependent to a significant degree on the size of the nuclear system served. For each required step, one or more concepts have been examined in detail to characterize the environmental impacts of construction, operation and decommissioning, the impacts of potential accidents, the dollar cost of construction and operation, and the safeguard requirements. Summary results of these evaluations are presented here. Detailed results are available in DOE/ET-0028 and DOE/ET-0029.

All of the concepts evaluated here are considered to represent available technology; that is, enough information is available to initiate design and construction of full-scale facilities, although varying degrees of design verification testing may be required. Brief descriptions are also provided of a number of alternative high-level waste treatment concepts that do not represent available technology but have attractive attributes that make them potential alternatives.

4.1 RELATIONSHIP OF PREDISPOSAL OPERATIONS TO DISPOSAL AND PROGRAM ALTERNATIVES

The relationships of the predisposal operations to the unique system requirements for each disposal alternative, for both the once-through and the fuel reprocessing cycles, are described in this section. The individual components of the predisposal systems are then described and analyzed in subsequent sections.

4.1.1 Predisposal System for the Once-Through Cycle

A simplified diagram of the predisposal waste management system for spent fuel in the once-through fuel cycle is shown in Figure 4.1.1. For the example predisposal system assumed here, the spent fuel is stored at the reactor storage basins for a minimum of 5 years. The fuel may be stored there for a longer period if a disposal facility is not available and if capacity is available at the reactor. The fuel is then shipped to a

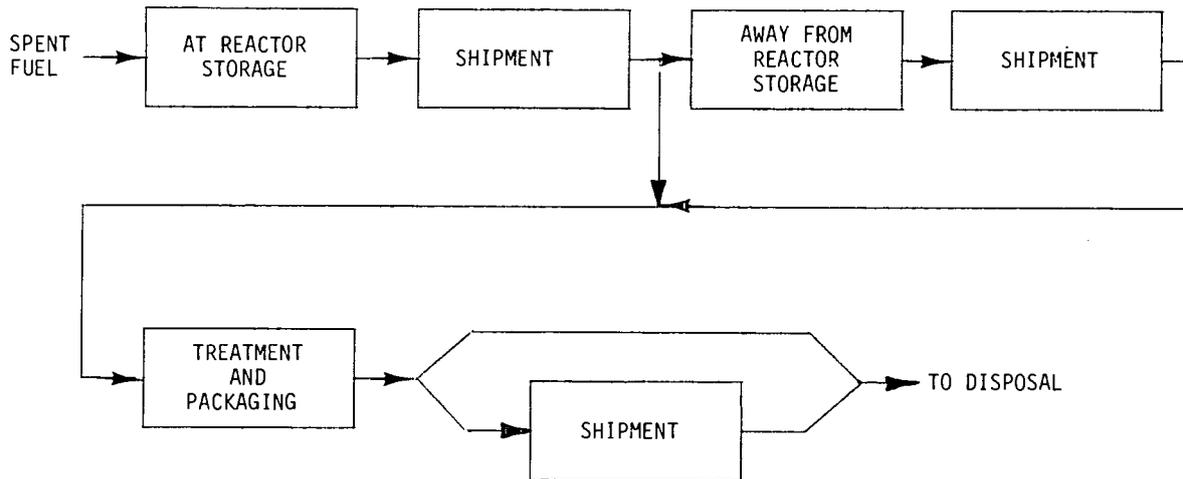


FIGURE 4.1.1 Predisposal Waste Management System for Spent Fuel in the Once-Through Fuel Cycle

treatment and packaging facility if a disposal facility is available. If a disposal facility is not available, the fuel is assumed to be shipped to an away-from-reactor (AFR) storage facility.^(a) When a disposal facility is available, the fuel is shipped there for treatment and packaging prior to disposal. Alternative approaches include having packaging facilities located separately from disposal facilities with extended storage of packaged fuel before disposal.

The types of operations and facilities considered in this Statement for each of the predisposal alternatives are identified in Table 4.1.1. This table shows that the initial storage and shipment operations are identical for all of the disposal alternatives. The differences in the predisposal systems are in the treatment and packaging and final shipment to disposal. Four of the eight alternatives to mined geologic disposal can utilize the same treatment and packaging options as mined geologic disposal; however, three of these require ocean ship transport to the final disposal site. Four of the alternatives can only be utilized in the once-through cycle if the spent fuel is first dissolved as in a reprocessing cycle. Two of these alternatives require disposal as liquid high-level waste. In these two cases, no shipment to disposal is required because the treatment facility and the disposal facility are located on a common site. The transmutation alternative requires, in addition to dissolution of the fuel, complex chemical partitioning, target fabrication, and irradiation. Space disposal requires, in addition to dissolution of the spent fuel, a process to convert the liquid waste into an encapsulated solid material. All of the alternatives that utilize a dissolution process would also generate considerable quantities of miscellaneous TRU waste. These would require the same treatment and handling as the comparable wastes produced in the reprocessing cycle described in the next subsection.

(a) AFR storage facilities were referred to as independent spent fuel storage facilities (ISFSFs) in DOE/ET-0028 and DOE/ET-0029.

TABLE 4.1.1. Predisposal Operations and Alternatives for Once-Through Cycle Disposal Options

<u>Disposal Option</u>	<u>Shipment to Interim Storage</u>	<u>Interim Storage</u>	<u>Shipment to Treatment</u>	<u>Treatment and Packaging</u>	<u>Shipment to Disposal</u>
Mined geologic	Rail and Truck	Water basin	Rail and truck	Encapsulate individual assemblies	None if onsite or rail if offsite
		<p>Alternatives include packaged fuel storage in:</p> <ul style="list-style-type: none"> ● Dry wells ● Air cooled vaults ● Surface casks 		<p>Alternatives include:</p> <ul style="list-style-type: none"> ● Encapsulate multiple assemblies ● Disassemble and encapsulate ● Chop, voloxidize and encapsulate ● Dissolve and convert to glass^(a) 	
Very deep holes	Same as above	Same as above	Same as above	Same as above	Same as above
Rock melting	Same as above	Same as above	Same as above	Dissolve and dispose as liquid ^(a,b)	Onsite disposal
Island	Same as above	Same as above	Same as above	Same as mined geologic island transports	Rail, ocean ship and island transporter
Subseabed	Same as above	Same as above	Same as above	Same as mined geologic	Rail and ocean ship
Ice sheet	Same as above	Same as above	Same as above	Same as mined geologic	Rail, ocean ship and over-ice vehicle
Well injection	Same as above	Same as above	Same as above	Dissolve and dispose as liquid ^(a,b)	Onsite disposal
Transmutation	Same as above	Same as above	Same as above	Dissolve, partition, fabricate targets, irradiate and reprocess targets ^(a)	Truck or rail to and from irradiation
Injection into Space	Same as above	Same as above	Same as above	Dissolve and convert to "cermet" matrix in capsules ^(a)	Rail to launch site; launch to orbit, see Section 6.1.8

(a) Spent fuel treatment involving dissolution produces TRU wastes requiring all of the TRU waste predisposal operations shown in Table 4.1.3. for reprocessing cycle wastes. These TRU wastes then probably will require mined geologic disposal.

(b) Disposal of spent fuel as an aqueous liquid in the rock melting and well injection options may not be feasible because of criticality questions.

4.1.2 Predisposal System for the Reprocessing Cycle

A simplified diagram of the predisposal waste management system for the reprocessing cycle is shown in Figure 4.1.2.^(a) In this cycle, wastes requiring disposal are produced at the fuel reprocessing plant (FRP) and at the mixed-oxide fuel fabrication plant (MOX-FFP). These wastes are assumed to be treated and packaged at the site where they are produced, either the FRP or MOX-FFP. They are then shipped to interim storage if a disposal facility is not available; finally, they are shipped to a disposal facility.

The operations and facilities required for the predisposal system for management of the high-level waste are shown in Table 4.1.2. As in the case of spent fuel, four of the alternatives to mined geologic disposal can utilize the same treatment and interim storage processes as the mined geologic disposal option. Three of the alternatives, however, require ocean transport to the final disposal site. In the two cases where high-level waste is disposed of as a liquid, the only predisposal system facilities required for high-level waste are the interim storage facilities consisting of double-walled below-grade tanks. For the transmutation alternative, interim storage is assumed to be required for the liquid high-level waste in double-walled below-grade tanks prior to the partitioning processing. This storage requirement and the target recycle requirements are thus exceptions to the sequence of operations shown in Figure 4.1.2. For space disposal, as in the once-through cycle, the high-level waste solution is converted to a solid "cermet" matrix contained in special spherical capsules. Interim storage would be similar to that of spent fuel, but because of the shape of the container, it would have its own unique design requirement.

Various TRU waste materials must also be disposed of in all of the disposal concepts. Although it may be possible to dispose of some of these materials after treatment in the same facility used for disposal of the high-level waste, it is assumed here that these materials are always sent to a mined geologic repository regardless of the disposal option selected for high-level waste. The operations and facilities considered for the predisposal system for these waste materials are shown in Table 4.1.3.

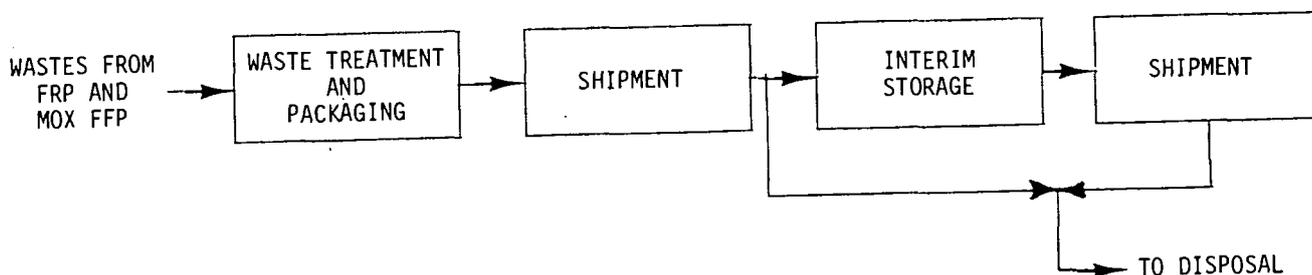


FIGURE 4.1.2. Predisposal Waste Management System for Fuel Reprocessing Plant and MOX-Fuel Fabrication Plant Wastes in the Fuel Reprocessing Cycle

(a) For a description of the fuel cycle prior to waste generation at the FRP and the MOX-FFP, see Figure 3.2.2.

TABLE 4.1.2. Predisposal Operations and Alternatives for Reprocessing-Cycle High-Level Liquid Wastes

<u>Disposal Option</u>	<u>Waste Treatment</u>	<u>Shipments to Interim Storage^(a)</u>	<u>Interim Storage</u>	<u>Shipment to Disposal</u>
Mined geologic	Convert to stable solid such as a calcine, a glass, ^(b) a synthetic mineral, a metal matrix, etc.	Rail ^(b) or truck	Water basins and/or air-cooled sealed casks ^(b)	Rail ^(b) or truck
Very deep holes	Same as above	Same as above	Same as above	Same as above
Rock melting	Not required	Not required	Double-walled tanks	Onsite disposal
Island	Same as mined geologic	Same as mined geologic	Same as mined geologic	Rail and ocean ship
Subseabed	Same as mined geologic	Same as mined geologic	Same as mined geologic	Rail and ocean ship
Ice sheet	Same as mined geologic	Same as mined geologic	Same as mined geologic	Rail and ocean ship
Well injection	Not required	Not required	Double walled tanks	Onsite disposal
Transmutation	Partition, fabricate targets, irradiate and reprocess targets	Not required	Double walled tanks	Truck or rail to and from irradiation
Injection into space	Convert to a "cermet" matrix in capsules	Same as mined geologic	Similar to mined geologic	Rail or truck to launch site; launch to orbit see Section 6.1.8

(a) A 5-year storage period in water basin facilities at the reprocessing plant is assumed before shipment to other interim storage.

(b) The example method of this Statement.

TABLE 4.1.3. Example Predisposal Operations and Alternatives Evaluated for Reprocessing-Cycle TRU Wastes for All Disposal Concepts

<u>Non-High-Level Waste Type</u>	<u>Waste Treatment</u>	<u>Shipments to Interim Storage</u>	<u>Interim Storage</u>	<u>Shipments To Disposal</u>
Fuel Residue(a)	Package in canisters without compaction.(b) Alternatives include: <ul style="list-style-type: none"> • Mechanical compaction of hulls • Hulls melting 	In casks by rail(b) or truck	Dry-well facility(b) or concrete vault	In casks by rail(b) or truck
Failed equipment and other non-combustible waste	Failed equipment decontaminated and disassembled as required. Non-combustible waste packaged without treatment. Packaged in canisters, drums and boxes	Canisters in casks by rail(b) or truck. High dose-rate drums in casks by rail or truck.(b) Other drums and boxes in shielded over-packs or special containers by rail or truck(b)	Canisters in dry-well facility(b) or concrete vaults. High dose-rate drums in dry-well facility or concrete vaults.(b) Low dose-rate containers in unshielded buildings or outdoors with earth cover(b)	Same as to interim storage
Combustible waste	Incinerate and immobilize ash in cement(b) or bitumen. Alternatives include packaging without treatment	Drums in casks or shielded over-packs or special containers by rail or truck(b)	High dose-rate drums in dry-well facility or concrete vaults.(b) Low dose-rate containers in unshielded buildings or outdoors with earth cover(b)	Same as to interim storage
Wet wastes and particulates	Immobilize in cement(b) or bitumen	Same as above	Same as above	Same as above
Gaseous and airborne wastes	Use high efficiency filters and process to remove I, C and Kr. Alternatives include ³ H removal	Recovered solids as above. ⁸⁵ Kr not shipped	⁸⁵ Kr stored on-site in special facility for pressurized gas cylinders. Other materials as above	Recovered solids as above. ⁸⁵ Kr not shipped off-site

- (a) Spent fuel cladding hulls and hardware that remain after fuel components have been leached out.
 (b) The example method of this Statement.

Although they are not necessarily waste management functions, the spent fuel handling and storage operations that occur before reprocessing are, to be conservative, also included in the predisposal system in the system simulation analyses in Chapter 7. This includes the operations shown in Figure 4.1.1 prior to treatment and packaging.

4.1.3 Predisposal System Relationships to Program Alternatives

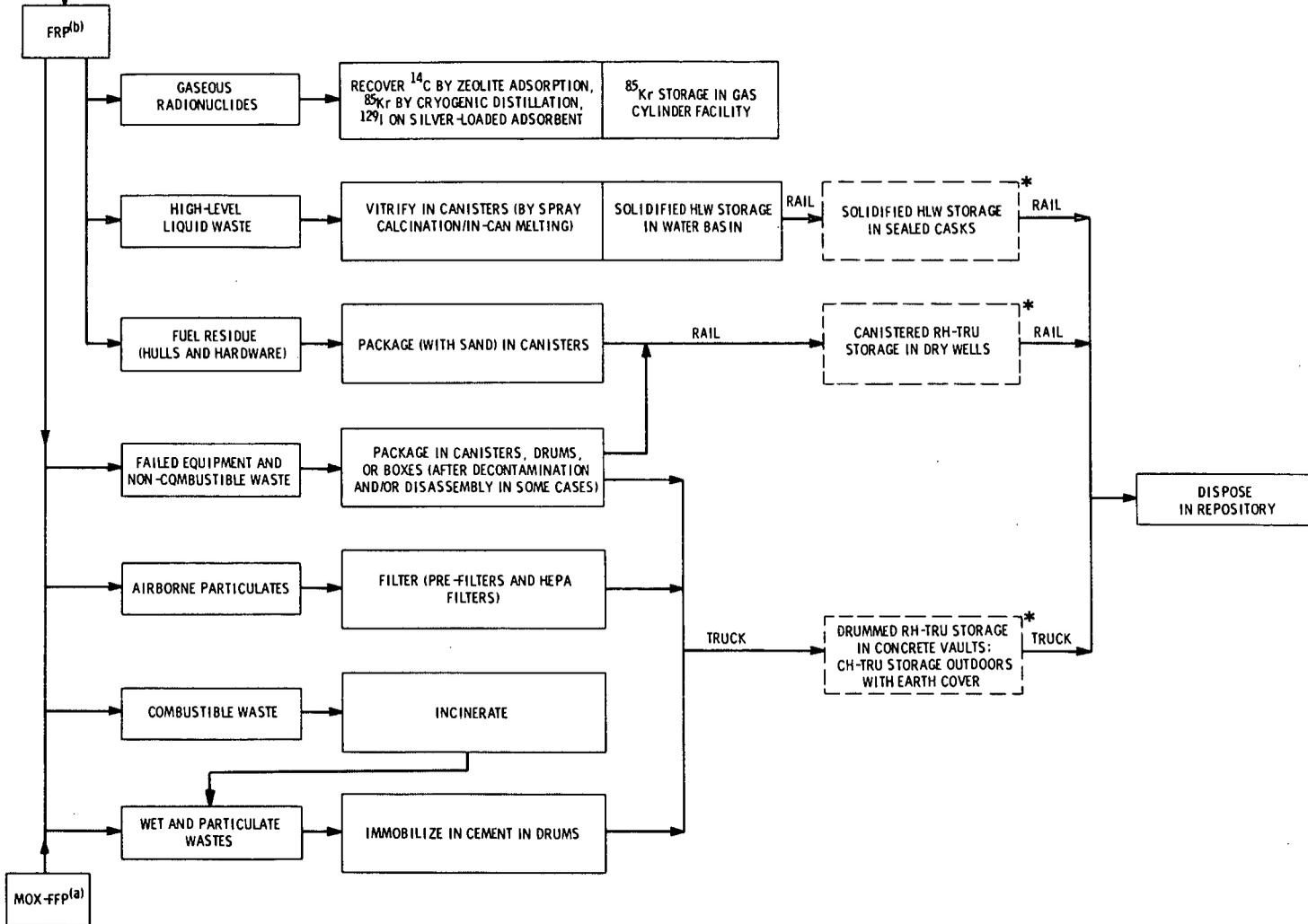
The predisposal systems for the preferred alternative, that is, a program leading to utilization of mined geological repositories, are listed with the mined geologic disposal option in Tables 4.1.1, 4.1.2 and 4.1.3. If the alternative program to develop several disposal options in parallel were to be adapted, some of the other predisposal operations shown in these tables might be utilized. For the no-action alternative, spent fuel would be stored indefinitely without either reprocessing or final disposal.

The predisposal waste management operations for the preferred alternatives are given schematically in more detail for both fuel cycles in Figure 4.1.3. These operations are discussed in more detail in Sections 4.3 to 4.6.

ONCE-THROUGH CYCLE



REPROCESSING CYCLE



* FUNCTIONS IN DASHED BOXES DEPEND ON REPOSITORY START-UP DATE (AND/OR REPROCESSING START-UP DATE, IN THAT CYCLE)

(a) EXAMPLE DECOMMISSIONING MODE IS IMMEDIATE DISMANTLEMENT

(b) EXAMPLE DECOMMISSIONING MODE IS 30-YR PASSIVE SAFE STORAGE BEFORE DISMANTLEMENT

FIGURE 4.1.3. Example Predisposal Waste Management Operations for the Mined Geologic Disposal Option

4.2 UNTREATED WASTE CHARACTERIZATION

The quantities and composition of the wastes generated at each step in the post-fission LWR fuel cycle have been studied in detail. Quantities used in this Statement are based upon actual practice for processes that have been demonstrated and upon technical judgments for processes that have not yet been commercially demonstrated. The untreated initial wastes, termed primary wastes, are identified, described, and classified as the first step in defining the environmental impact of radioactive waste treatment. Additional details are presented in DOE/ET-0028 (Section 3.3).

The primary wastes are processed to form treated wastes suitable for disposal. It is anticipated that essentially all commercial wastes (on a Curie basis) or a large fraction (on a volume basis) will receive treatment. Treated wastes are of two types: 1) gaseous wastes that have been treated to reduce their activity levels so they can be released to the environment without harm to man, and 2) wastes that have been converted to a stable form suitable for disposal so that their radioactivity will remain confined and out of contact with man's environment.

Secondary wastes are generated in the treatment of primary wastes and in the subsequent handling of treated wastes. Secondary wastes are generated not only from initial waste processing, but also from the storage, transportation, and isolation steps. In most cases, the amount of secondary wastes is small in comparison to the amount of primary wastes; nevertheless, an assessment of the environmental impacts is not complete without including the effects of the secondary wastes. Treated secondary wastes are included with the treated primary wastes in Section 4.3.7.

Decommissioning wastes result from the operations employed to decommission retired nuclear fuel cycle facilities. These wastes must also be included in a complete analysis of the impacts of nuclear waste treatment; characterization of such wastes is presented in Section 4.6.

Many methods of classifying radioactive wastes are in use, based on the kind of radioactivity contained, the amount of radioactivity contained, the untreated physical form, the treated physical form, etc. In this Statement, wastes have been classified into categories based on their treatment requirement; i.e., all wastes requiring a similar treatment are included in the same category. The categories and a brief generic description of each are given in Table 4.2.1. The first three waste categories are specific to certain fuel cycles. Spent fuel as a waste is specific only to the once-through cycle, and high-level liquid waste and fuel residue are specific only to the reprocessing cycle. The last four waste categories listed in Table 4.2.1 are generated in almost every facility in which radioactive materials are processed, treated, or handled. Thus, both primary and secondary wastes of these categories are found throughout the LWR fuel cycles.

Radioactive wastes are also generally classified according to their content of transuranic (TRU) radionuclides (i.e., radionuclides with atomic number greater than 92). Because of the long half-lives and high radiotoxicity of some TRU nuclides, TRU wastes are

TABLE 4.2.1. Classification of Primary Wastes from the Post-Fission LWR Fuel Cycle

Waste Category	General Description
Spent fuel	Irradiated PWR and BWR fuel assemblies containing fission products and actinides in ceramic UO ₂ pellets sealed in Zircaloy tubes. Intense radioactivity.
High-level liquid waste	Contains about 0.5% of the U and Pu in the spent fuel and over 99% of the fission products and other actinides. Intense radioactivity.
Fuel residue	Includes short segments of Zircaloy tubing (hulls) remaining after UO ₂ is dissolved and stainless steel assembly hardware.
Gaseous	Predominately two types: 1) large volumes of ventilation air, potentially containing particulate activity, and 2) smaller volumes of vessel vent and process off-gas, potentially containing volatile radioisotopes in addition to particulate activity.
Compactable and combustible wastes	Miscellaneous wastes including paper, cloth, plastic, rubber, and filters. Wide range of radiation levels dependent on source of waste.
Concentrated liquids, wet wastes, and particulate solids.	Miscellaneous wastes including evaporator bottoms, filter sludges, resins, etc. Wide range of radioactivity levels dependent on source of waste.
Failed equipment and noncombustible wastes	Miscellaneous metal or glass wastes including massive process vessels. Wide range of radioactivity levels dependent on source of waste.

considered more hazardous than non-TRU wastes. Present regulations governing disposal of TRU wastes are more stringent than those governing disposal of non-TRU wastes. Non-TRU wastes are eligible for disposal by surface burial and, except for gaseous and airborne wastes, some of which contain non-TRU radionuclides of special concern (¹²⁹I, ⁸⁵Kr and ¹⁴C), management of these wastes is outside the scope of this Statement. However, data on the characteristics of untreated post-fission non-TRU wastes are included in DOE/ET-0028 (Section 3.3) along with those of the TRU wastes.

In current practice, a TRU waste is considered to be one that contains more than 10 nanocuries of transuranic alpha activity per gram of waste. However, spent fuel as waste and high-level waste that results from processing spent fuel, which contain high levels of transuranic activity, are considered as a separate high-level waste category. Raising the dividing line between TRU and non-TRU wastes from 10 nCi/g to 100 nCi/g has been proposed by EPA. Because these low concentrations are often difficult to measure in wastes, we assume in this Statement that all wastes from locations that might cause contamination levels above 10 nCi/g of waste are considered to be TRU-suspect and are combined with known TRU wastes for treatment.

In order to relate waste quantities to electric energy generation and to facilitate comparisons between alternative nuclear fuel cycles, the waste volumes and activities in this section are given per GWe-yr. One GWe-yr (or 8.8×10^9 kWh) is equivalent to the annual output of one of the largest nuclear power plants operating today (a 1250 MWe plant operating for one year at 80% capacity produces 1 GWe-yr of electricity). One GWe-yr also

corresponds to the annual electrical energy consumption of about one million people in the U.S. (The total electric utility sales in 1978 amounted to about 230 GWe-yr.) For the generic LWR fuel cycle upon which this Statement is based, 38 MT of UO_2 or mixed UO_2 - PuO_2 (MOX) fuel must pass through the cycle to generate 1 GWe-yr.

4.2.1 Once Through-Cycle Wastes

The only primary waste in the once-through fuel cycle within the scope of this Statement is the spent fuel itself. Two basic types of LWR fuel are in use today: pressurized water reactor (PWR) fuel and boiling water reactor (BWR) fuel. The reference PWR and BWR fuel assemblies defined for this generic Statement are described in Figure 4.2.1. Fuel for specific plants may vary somewhat from these descriptions.

For the purpose of describing radioactivity content of the wastes here, an example fuel composition based on a representative mixture of PWR and BWR fuel assemblies was developed. However, the system simulation results presented in Chapter 7 are based on explicit PWR and

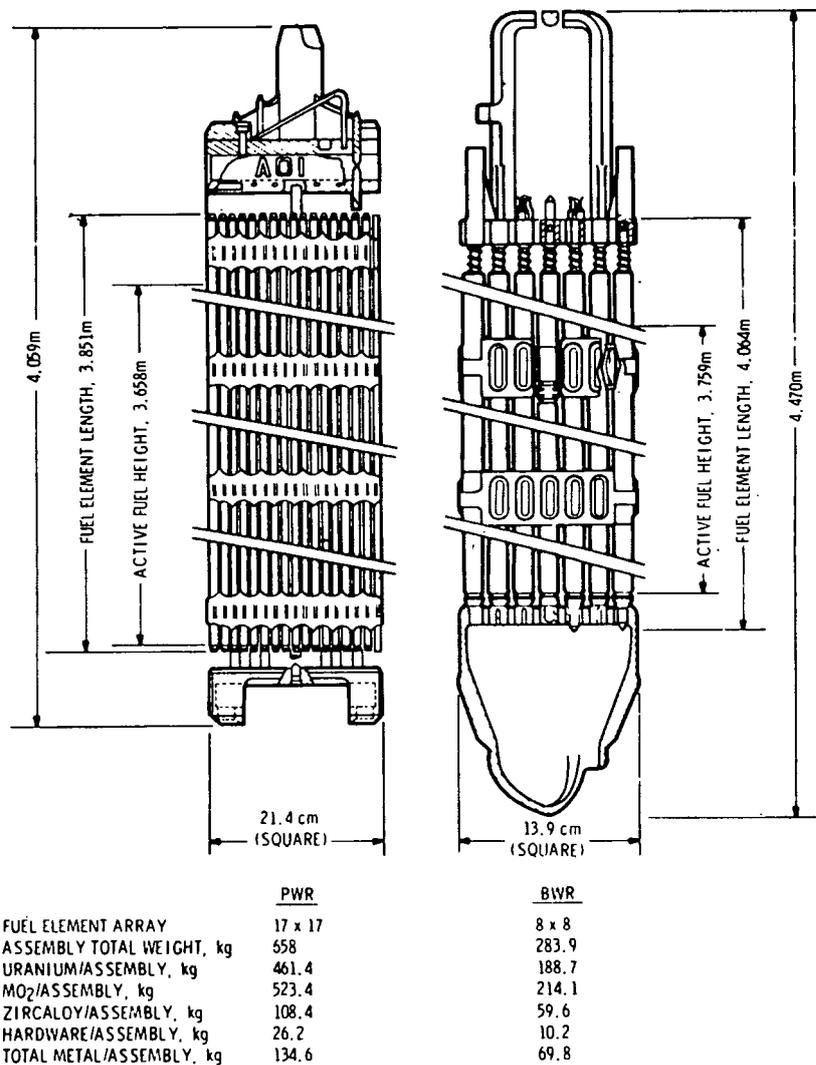


FIGURE 4.2.1. Unirradiated Reference Fuel Assemblies

BWR fuel models that account for all radionuclides in the fuel and take into account differences in fuel exposures for PWR and BWR fuel assemblies and the effects of reduced exposure for startup and shutdown cores.

The amounts of some selected radionuclides present in the example fuel composition are listed in Table 4.2.2. These radionuclides were selected based on several factors, among which are 1) potential for release, 2) potential effect of release, 3) quantity present, and 4) public interest. These nuclides and their radioactive daughter nuclides provide most of

TABLE 4.2.2. Selected Radionuclide Content in Example Once-Through Cycle Spent Fuel

Fission Products (a)	Ci/GWe-yr for Various Decay Periods				
	1.5 yr	5 yr ^(b)	10 yr	50 yr	100 yr
³ H (1.2 x 10 ¹)	1.6 x 10 ⁴	1.3 x 10 ⁴	9.5 x 10 ³	1.0 x 10 ³	6.1 x 10 ¹
⁸⁵ Kr (1.1 x 10 ¹)	3.4 x 10 ⁵	2.7 x 10 ⁵	1.9 x 10 ⁵	1.5 x 10 ⁴	6.1 x 10 ²
⁹⁰ Sr (2.9 x 10 ¹)	2.5 x 10 ⁶	2.2 x 10 ⁶	2.0 x 10 ⁶	7.4 x 10 ⁵	2.2 x 10 ⁵
¹⁰⁶ Ru (1.0)	6.5 x 10 ⁶	3.8 x 10 ⁵	1.3 x 10 ⁴		
¹²⁹ I (1.6 x 10 ⁷)	1.3	1.3	1.3	1.3	1.3
¹³⁴ Cs (2.1)	4.6 x 10 ⁶	1.2 x 10 ⁶	2.2 x 10 ⁵	2.9 x 10 ⁻¹	
¹³⁷ Cs (3.0 x 10 ¹)	3.5 x 10 ⁶	3.2 x 10 ⁶	2.9 x 10 ⁶	1.1 x 10 ⁶	3.6 x 10 ⁵
¹⁴⁴ Ce (7.8 x 10 ⁻¹)	9.5 x 10 ⁶	2.7 x 10 ⁵	3.1 x 10 ³		
Total all Fission Products	5.3 x 10 ⁷	1.6 x 10 ⁷	1.0 x 10 ⁷	3.7 x 10 ⁶	1.1 x 10 ⁶
Actinides (a)					
²³⁸ U (4.5 x 10 ⁹)	1.2 x 10 ¹	1.2 x 10 ¹	1.2 x 10 ¹	1.2 x 10 ¹	1.2 x 10 ¹
²³⁸ Pu (8.9 x 10 ¹)	8.0 x 10 ⁴	7.9 x 10 ⁴	7.6 x 10 ⁴	5.6 x 10 ⁴	3.8 x 10 ⁴
²³⁹ Pu (2.4 x 10 ⁴)	1.1 x 10 ⁴	1.1 x 10 ⁴	1.1 x 10 ⁴	1.1 x 10 ⁴	1.1 x 10 ⁴
²⁴⁰ Pu (6.8 x 10 ³)	1.7 x 10 ⁴	1.7 x 10 ⁴	1.7 x 10 ⁴	1.7 x 10 ⁴	1.7 x 10 ⁴
²⁴¹ Pu (1.3 x 10 ¹)	4.2 x 10 ⁶	3.4 x 10 ⁶	2.6 x 10 ⁶	4.1 x 10 ⁵	3.8 x 10 ³
²⁴¹ Am (4.6 x 10 ²)	1.4 x 10 ⁴	3.5 x 10 ⁴	6.1 x 10 ⁴	1.3 x 10 ⁵	1.3 x 10 ⁵
²⁴² Cm (4.5 x 10 ⁻¹)	1.4 x 10 ⁵	6.0 x 10 ²	3.2 x 10 ²	2.7 x 10 ²	2.1 x 10 ²
²⁴⁴ Cm (1.8 x 10 ¹)	4.9 x 10 ⁴	4.2 x 10 ⁴	3.4 x 10 ⁴	7.4 x 10 ³	1.1 x 10 ³
Total All Actinides	4.5 x 10 ⁶	3.6 x 10 ⁶	2.9 x 10 ⁶	6.3 x 10 ⁵	2.4 x 10 ⁵
Activation Products (a)					
¹⁴ C (5.7 x 10 ³)	2.8 x 10 ¹	2.8 x 10 ¹	2.8 x 10 ¹	2.8 x 10 ¹	2.8 x 10 ¹
⁵⁵ Fe (2.4)	1.6 x 10 ⁵	3.8 x 10 ⁴	1.3 x 10 ⁴	2.7 x 10 ⁻¹	4.5 x 10 ⁻⁷
⁶⁰ Co (5.3)	1.6 x 10 ⁵	1.1 x 10 ⁵	4.0 x 10 ⁴	2.6 x 10 ³	2.8 x 10 ⁻¹
⁶³ Ni (9.2 x 10 ¹)	1.5 x 10 ⁴	1.5 x 10 ⁴	1.5 x 10 ⁴	1.3 x 10 ⁴	7.7 x 10 ³
Total All Activation Products	3.5 x 10 ⁵	2.1 x 10 ⁵	7.2 x 10 ⁴	1.6 x 10 ⁴	7.7 x 10 ³

(a) Numbers in parentheses are the half-lives (in years).

(b) A minimum age of 5 yr is assumed here for shipment of spent fuel from the reactors in the once-through cycle.

the radioactivity in spent fuel while predisposal operations take place. Tables in Appendix A of Volume 2 provide data for these and other radionuclides for longer time periods.

Substantial quantities of non-TRU wastes are generated in the once-through fuel cycle during operation of nuclear power plants and spent fuel storage facilities. Depending on the treatment in the once-through fuel cycle, substantial amounts of TRU secondary wastes may or may not be produced. If the treatment mode involves simply the packaging of intact spent fuel, the secondary waste produced in the packaging operation should contain very little TRU radioactivity and is considered here to be all non-TRU waste. However, if the spent fuel cladding is breached in the treatment process, then secondary TRU wastes would be produced. Depending on the complexity of such a process, substantial amounts of TRU secondary waste could be produced. The secondary TRU wastes from the once-through fuel cycle would be similar to some of the primary wastes in the reprocessing case.

4.2.2 Reprocessing Cycle

When spent fuel is processed to recover (for recycle) the uranium and plutonium it contains, primary TRU wastes of two types are generated in recycle facilities: 1) fuel reprocessing plant (FRP) wastes and 2) mixed oxide fuel fabrication plant (MOX-FFP) wastes. In fuel reprocessing plants the spent fuel is dissolved out of the cladding, the uranium and plutonium are recovered and purified by a series of solvent extraction operations, and the uranium and plutonium products are converted to UF_6 and PuO_2 (or mixed UO_2 - PuO_2) for further use. In mixed oxide fuel fabrication plants the PuO_2 (or mixed UO_2 - PuO_2) is blended with UO_2 , processed to a suitable form, and incorporated into mixed oxide (MOX) fuel elements to be recycled to a nuclear power plant. More extensive descriptions of such facilities are presented in DOE/ET-0028 (Section 3.2).

Table 4.2.3 contains the estimated quantities and selected radionuclide contents of the primary high-level, TRU, and gaseous wastes generated in the reprocessing cycle. The radionuclide contents are given as fractions of the amounts present in the recycle spent fuel for the FRP wastes and as fractions of the amounts present in the fabricated MOX fuel for the MOX FFP wastes. These amounts are presented in Table 4.2.4, for an example recycle spent fuel, and in Table 4.2.5, for an example MOX fuel. Except for the isotopes of uranium and plutonium, the total amounts of radionuclides present in the untreated wastes of the two fuel cycles may be directly compared using the data of Tables 4.2.2 and 4.2.4. The quantities of uranium and plutonium in the reprocessing cycle wastes amount to about 1% of that present in the spent fuel.

Wastes from two areas of the fuel reprocessing plant (the fuel storage basin and the uranium conversion facility) are classified as non-TRU wastes. As in the once-through cycle, non-TRU wastes also result from operation of nuclear power plants and spent fuel storage facilities.

TABLE 4.2.3. Selected Radionuclide Content in Primary High-Level, TRU, and Gaseous Wastes from Fuel Reprocessing Plant and MOX Fuel Fabrication Plant

Waste Category	Facility	Volume, (a) m ³ /GWe-yr	Nuclide Content in Waste Category(b) as a Fraction of that Present in Spent Fuel (c) or in MOX Fuel(d)												
			Fission Products						Actinides			Activation Products			
			H	Kr	Sr, Cs	Ru	I	Ce	Pu	Am	Cm	C	Fe	Co	Ni
High-Level Liquid Waste	FRP	22	0.08	0	1	1	5 x 10 ⁻³	1	5 x 10 ⁻³	1	1	0	0	0	0
Fuel Residue	FRP														
Hulls		10	0.15	0	5 x 10 ⁻⁴	5 x 10 ⁻⁴	0	5 x 10 ⁻⁴	5 x 10 ⁻⁴	5 x 10 ⁻⁴	5 x 10 ⁻⁴	0.09	0.02	0.04	0.01
Hardware		2.1	0	0	0	0	0	0	0	0	0	0.07	0.98	0.96	0.99
Failed Equipment	FRP	8.4	0	0	1 x 10 ⁻⁶	1 x 10 ⁻⁶	0	1 x 10 ⁻⁶	1 x 10 ⁻⁴	1 x 10 ⁻⁶	1 x 10 ⁻⁶	0	0	0	0
	MOX FFP	1.5	0	0	0	0	0	0	1 x 10 ⁻⁵ (d)	1 x 10 ⁻⁶ (d)	1 x 10 ⁻⁶	0	0	0	0
Noncombustible Waste	FRP	15	0	0	1 x 10 ⁻⁶	1 x 10 ⁻⁶	0	1 x 10 ⁻⁶	1 x 10 ⁻⁴	1 x 10 ⁻⁶	1 x 10 ⁻⁶	0	0	0	0
	MOX FFP	1.5	0	0	0	0	0	0	1 x 10 ⁻⁴ (d)	1 x 10 ⁻⁴ (d)	0	0	0	0	0
Compactable and Combustible Waste															
Trash + Process Matl's	FRP	62	0	0	1.1 x 10 ⁻⁶	1.1 x 10 ⁻⁵	2.1 x 10 ⁻³	1.1 x 10 ⁻⁶	6.1 x 10 ⁻⁴	1.1 x 10 ⁻⁶	1.1 x 10 ⁻⁶	0	0	0	0
	MOX FFP	3.8	0	0	0	0	0	0	3 x 10 ⁻⁴ (d)	3 x 10 ⁻⁴ (d)	0	0	0	0	0
Filters	FRP	6.1	0	0	1 x 10 ⁻⁵	1 x 10 ⁻⁵	0	1 x 10 ⁻⁵	2 x 10 ⁻³	1 x 10 ⁻⁵	1 x 10 ⁻⁵	0	0	0	0
	MOX FFP	0.76	0	0	0	0	0	0	7 x 10 ⁻⁴ (d)	7 x 10 ⁻⁴ (d)	0	0	0	0	0
Concentrated Liquids, Wet Wastes, and Particulate Solids	FRP	5.7	1 x 10 ⁻³	0	1 x 10 ⁻⁵	1 x 10 ⁻³	3 x 10 ⁻³	1 x 10 ⁻⁵	1 x 10 ⁻³	1 x 10 ⁻⁵	1 x 10 ⁻⁵	0	0	0	0
	MOX FFP	2.8	0	0	0	0	0	0	1.1 x 10 ⁻⁴ (d)	2 x 10 ⁻² (d)	0	0	0	0	0
Gaseous Wastes															
Dissolver Off-Gas	FRP	2.6 x 10 ⁴ (b)	0.05	1	1 x 10 ⁻⁷	2 x 10 ⁻⁴	1	1 x 10 ⁻⁷	1 x 10 ⁻⁷	1 x 10 ⁻⁷	1 x 10 ⁻⁷	0.84	0	0	0
Vessel Off-Gas	FRP	1.1 x 10 ⁶ (b)	1 x 10 ⁻³	1 x 10 ⁻⁶	1 x 10 ⁻⁷	1 x 10 ⁻⁷	5 x 10 ⁻³	1 x 10 ⁻⁷	1 x 10 ⁻⁷	1 x 10 ⁻⁷	1 x 10 ⁻⁷	0	0	0	0
Vaporized Excess Water	FRP	7.6 x 10 ⁵ (b)	0.72	1 x 10 ⁻¹⁰	1 x 10 ⁻¹⁶	1 x 10 ⁻¹⁰	1 x 10 ⁻⁵	1 x 10 ⁻¹⁶	1 x 10 ⁻¹¹	1 x 10 ⁻¹⁶	1 x 10 ⁻¹⁶	0	0	0	0
	MOX FFP	1.5 x 10 ⁴ (b)	0	0	0	0	0	0	1 x 10 ⁻¹² (d)	1 x 10 ⁻¹⁰ (d)	0	0	0	0	0
Ventilation Air	FRP	1.2 x 10 ⁸ (b)	1 x 10 ⁻¹¹	1 x 10 ⁻¹¹	1 x 10 ⁻¹¹	1 x 10 ⁻¹¹	1 x 10 ⁻¹¹	1 x 10 ⁻¹¹	1 x 10 ⁻¹¹	1 x 10 ⁻¹¹	1 x 10 ⁻¹¹	1 x 10 ⁻¹¹	0	0	0
	MOX FFP	8.4 x 10 ⁶ (b)	0	0	0	0	0	0	3 x 10 ⁻⁷ (d)	3 x 10 ⁻⁷ (d)	0	0	0	0	0

(a) Data obtained from Section 4 of DOE/ET-0028.

(b) Data obtained from Section 3.3 of DOE-ET-0028.

(c) Quantities present in spent fuel are listed in Table 4.2.4.

(d) Quantities present in MOX fuel are listed in Table 4.2.5.

TABLE 4.2.4. Selected Radionuclide Content in Example Recycle Spent Fuel

Fission Products	Ci/GWe-yr for Various Decay Periods				
	1.5 yr ^(a)	6.5 yr ^(b)	10 yr	50 yr	100 yr
³ H	1.6 x 10 ⁴	1.2 x 10 ⁴	9.9 x 10 ³	1.0 x 10 ³	6.1 x 10 ¹
⁸⁵ Kr	3.2 x 10 ⁵	2.3 x 10 ⁵	1.8 x 10 ⁵	1.4 x 10 ⁴	5.7 x 10 ²
⁹⁰ Sr	2.3 x 10 ⁶	2.2 x 10 ⁶	1.9 x 10 ⁶	6.9 x 10 ⁵	2.0 x 10 ⁵
¹⁰⁶ Ru	7.2 x 10 ⁶	2.2 x 10 ⁵	1.4 x 10 ⁴		
¹²⁹ I	1.3	1.3	1.3	1.3	1.3
¹³⁴ Cs	4.6 x 10 ⁶	8.4 x 10 ⁵	2.2 x 10 ⁵	2.9 x 10 ⁻¹	
¹³⁷ Cs	3.5 x 10 ⁶	3.2 x 10 ⁶	2.9 x 10 ⁶	1.1 x 10 ⁶	1.1 x 10 ⁶
¹⁴⁴ Ce	9.1 x 10 ⁶	1.1 x 10 ⁵	3.0 x 10 ³		
Total All Fission Products	5.3 x 10 ⁷	1.3 x 10 ⁷	1.0 x 10 ⁷	3.6 x 10 ⁶	1.1 x 10 ⁶
<u>Actinides</u>					
²³⁸ U	1.2 x 10 ¹	1.2 x 10 ¹	1.2 x 10 ¹	1.2 x 10 ¹	1.2 x 10 ¹
²³⁸ Pu	2.1 x 10 ⁵	2.1 x 10 ⁵	2.0 x 10 ⁵	1.5 x 10 ⁵	9.9 x 10 ⁴
²³⁹ Pu	1.4 x 10 ⁴	1.4 x 10 ⁴	1.4 x 10 ⁴	1.4 x 10 ⁴	1.4 x 10 ⁴
²⁴⁰ Pu	2.8 x 10 ⁴	2.8 x 10 ⁴	2.8 x 10 ⁴	2.8 x 10 ⁴	2.8 x 10 ⁴
²⁴¹ Pu	6.8 x 10 ⁶	5.3 x 10 ⁶	4.6 x 10 ⁶	6.7 x 10 ⁵	6.5 x 10 ⁴
²⁴¹ Am	2.7 x 10 ⁴	7.6 x 10 ⁴	1.0 x 10 ⁵	2.2 x 10 ⁵	2.2 x 10 ⁵
²⁴² Cm	3.8 x 10 ⁵	1.6 x 10 ³	1.4 x 10 ³	1.2 x 10 ³	9.5 x 10 ²
²⁴⁴ Cm	2.7 x 10 ⁵	2.2 x 10 ⁵	1.9 x 10 ⁵	4.2 x 10 ⁴	6.1 x 10 ³
Total All Actinides	7.6 x 10 ⁶	5.7 x 10 ⁶	5.1 x 10 ⁶	1.1 x 10 ⁶	4.6 x 10 ⁵
<u>Activation Products</u>					
¹⁴ C	2.1 x 10 ¹	2.1 x 10 ¹	2.1 x 10 ¹	2.1 x 10 ¹	2.1 x 10 ¹
⁵⁵ Fe	1.6 x 10 ⁵	3.8 x 10 ⁴	1.3 x 10 ⁴	2.7 x 10 ⁻¹	4.5 x 10 ⁻⁷
⁶⁰ Co	1.6 x 10 ⁵	8.0 x 10 ⁴	4.0 x 10 ⁴	2.6 x 10 ³	2.8 x 10 ⁻¹
⁶³ Ni	1.5 x 10 ⁴	1.5 x 10 ⁴	1.5 x 10 ⁴	1.3 x 10 ⁴	7.7 x 10 ³
Total All Activation Products	3.5 x 10 ⁵	1.3 x 10 ⁵	7.2 x 10 ⁴	1.3 x 10 ⁴	7.7 x 10 ³

(a) A minimum age of 1.5 yr is assumed here for reprocessing.

(b) A minimum age of 6.5 yr is assumed here for shipment of solidified high-level waste.

TABLE 4.2.5. Selected Radionuclide Content in Example MOX fuel

Actinides	Ci/GWe-yr ^(a) for Different Times Since Reprocessing	
	1 yr ^(b)	10 yr
²³⁸ Pu	1.4×10^5	1.4×10^5
²³⁹ Pu	9.9×10^3	9.9×10^3
²⁴⁰ Pu	2.0×10^4	2.0×10^4
²⁴¹ Pu	4.4×10^6	2.9×10^6
²⁴¹ Am	7.3×10^3	5.9×10^4
Total	4.6×10^6	3.1×10^6

(a) Assuming 20% of fuel to reactors is recycle MOX fuel.

(b) A period of 1 yr is assumed here between reprocessing and MOX fuel fabrication.

4.3 WASTE TREATMENT AND PACKAGING

This section addresses the treatment and packaging of high-level (including spent fuel), TRU, and gaseous wastes resulting from the once-through and the reprocessing cycles. The principal source of the information contained herein is DOE/ET-0028, Technology for Commercial Radioactive Waste Management (DOE 1979), which was prepared in support of this Statement.^(a) The processes described here are not necessarily optimized but are representative of currently available technology.

The treated waste form and container each provide a barrier to release of radionuclides after disposal. The functions of the treated waste forms and containers are discussed in more detail in Section 5.1.2.

4.3.1 Spent Fuel Treatment and Packaging in Once-Through Cycle

In the once-through fuel cycle, the spent fuel is considered to be waste and is treated to prepare it for disposal. Treatment processes that have been examined range from simply 1) packaging the intact spent fuel assemblies to 2) chopping the fuel assemblies to expose the fuel, utilizing a process called voloxidation to remove a portion of the volatile radionuclides, dissolving the fuel in nitric acid and finally converting the solution to a solid by calcination and vitrification.

Encapsulation of intact spent fuel assemblies for geologic disposal is the example process assumed in this Statement for the once-through fuel cycle. Three other treatment methods are also described to illustrate the range of treatment alternatives available.

4.3.1.1 Encapsulate Intact Assembly (Example Method)

A detailed description of the example encapsulation process is contained in DOE/ET-0028 (Section 5.7.3). A similar process is described in ONWI-39 (Appendix C). In both of these process concepts the intact fuel assemblies are placed in steel canisters that are then backfilled with helium and welded closed. A flow diagram for the process is shown in Figure 4.3.1.

The canister and filler materials included in the studies discussed here are only a few of the potentially applicable materials. Canister materials being considered by DOE include a variety of metal alloys, ceramics, carbides, forms of carbon, glasses, and cements; potential filler (stabilizer) materials include a variety of gases, castable solids, and granular

(a) Additional once-through cycle concepts were discussed later in "An Assessment of LWR Spent Fuel Disposal Options," ONWI-39 (ONWI 1979); this report also contains information on a reprocessing case which is somewhat different in waste treatment philosophy than that presented in DOE/ET-0028. Other recent descriptions of reprocessing waste treatment operations are contained in "Design Integration Study, Spent LWR Fuel Recycle Complex, Conceptual Design, Case A-1, Separated Streams," DP-CFP-78-121 (SRL 1978) and "Design Integration Study, Spent LWR Fuel Recycle Complex, Conceptual Design, Coprocessing Case A-2," DP-CFP-121-79 (Harries et al. 1979). Various methods of waste treatment and packaging for both fuel cycles are also addressed in "Technical Support of Standards for High-Level Radioactive Waste Management, Volume B, Engineering Controls," EPA 520/4-79-007B (EPA 1977).

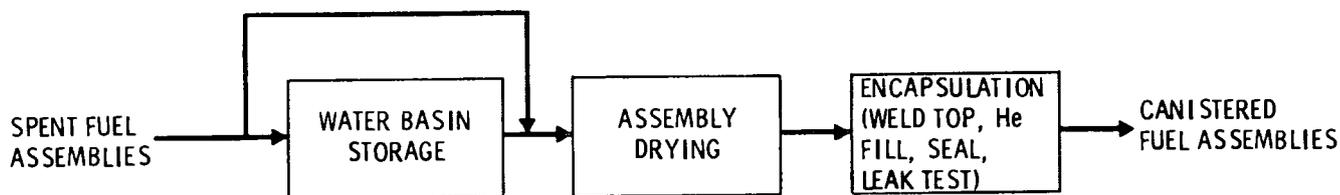


FIGURE 4.3.1. Flow Diagram for Encapsulation of Intact Spent Fuel Assemblies

solids (DOE/NE-0007, Section II.E.1). The waste package finally chosen will be tailored to the geologic environment in which the package is to be disposed.

In the DOE/ET-0028 study, the cleaned and dried fuel assemblies are individually packaged in square canisters^(a) that are only slightly larger than the assemblies themselves. A canister for a PWR assembly has dimensions of 0.24 x 0.24 x 4.88 m (9.5 x 9.5 x 192 in.) and a canister for a BWR assembly has dimensions of 0.165 x 0.165 x 4.88 m (6.5 x 6.5 x 192 in.). For the mixture of fuel used in this generic study (40% of the assemblies are from PWRs and 60% are from BWRs), 127 canisters are filled per GWe-yr.

The process concept described in ONWI-39 (Appendix C) is very similar except that cylindrical canisters are used, and the BWR assemblies are packaged three to a canister. A canister for a PWR assembly has dimensions of 0.36 x 4.72 m (14 x 186 in.) and a canister for three BWR assemblies has dimensions of 0.41 x 4.72 m (16 x 186 in.). Seventy-eight canisters per GWe-yr are required in this instance for the mixture of fuel used in this generic study.

The DOE/ET-0028 and the ONWI-39 studies present different estimates of TRU waste produced during the treatment operations. DOE/ET-0028 concluded that waste produced during the treatment of the intact fuel assemblies could be considered to be non-TRU (as is waste produced during the irradiation and the subsequent storage of the assemblies). ONWI-39, however, lists appreciable quantities of TRU wastes resulting from packaging of the intact assemblies (but does not say in which operations they arise). The actual amount remains to be determined from operating experience; if a significant amount of TRU waste is indeed generated during the packaging of intact spent fuel, then the spent fuel capacity of the repositories described in Chapter 5 may be somewhat overstated.

Consideration is also given in ONWI-39 (Section 10.3) to other canister design variations. Alternative canister void filler materials considered include gases other than helium (e.g., air, nitrogen, or argon), monolithic solid fillers formed by pouring molten materials (e.g., lead, aluminum, or glass) into the canister and then cooling, and granular solid fillers (e.g., lead shot, sand, or glass frit). The use of thicker walls in the primary canisters, overpacks, and increasing the number of spent fuel assemblies per canister were also considered.

(a) Square canisters allow a more close-packed array during interim storage but are not as strong as cylindrical canisters with the same wall thickness.

Another variation considered in ONWI-39 (Section 10.6) involves disassembly prior to packaging so that the canisters contain spent fuel rods only, instead of complete assemblies. In this option the end fittings are removed from the fuel elements, the elements are disassembled, and the fuel rods are bundled together and sealed into canisters.

4.3.1.2 Chop Fuel Assembly, Voloxidize Fuel, and Encapsulate

A process for chopping the fuel assemblies, removing volatile components through voloxidation, and encapsulating the spent fuel is described in ONWI-39 (Appendix C). The end fittings of the spent fuel are first cut off and encapsulated. The remaining portions of the fuel assemblies are then chopped and voloxidized, and encapsulated in canisters. A flow diagram for the process is shown in Figure 4.3.2.

The voloxidation process, which is in the development stage (Groenier 1977), promotes the release of gaseous fission products from the fuel by oxidizing UO_2 to U_3O_8 at 400° to $500^\circ C$ in air. This oxidation results in disintegration of the fuel, which provides an easier escape path for the gaseous fission products. Removal of the gaseous fission products from the off-gas stream is addressed in Section 4.3.4.

The processed spent fuel is encapsulated in cylindrical steel canisters that are helium-filled, sealed by welding, and leak tested. Any leaking canisters are overpacked in a second larger canister. The primary canister size is 0.30×3.0 m (12×120 in.). Sixty-one canisters per GWe-yr are estimated to be required to contain the chopped and voloxidized fuel.

The end fittings sheared from the fuel-bearing portions of spent fuel are packaged without further processing in 0.5×3.0 m cylindrical canisters. One canister holds the ends of either three PWR or six BWR assemblies; for the mixture of fuel used in this generic study, 11.6 canisters are filled per GWe-yr.

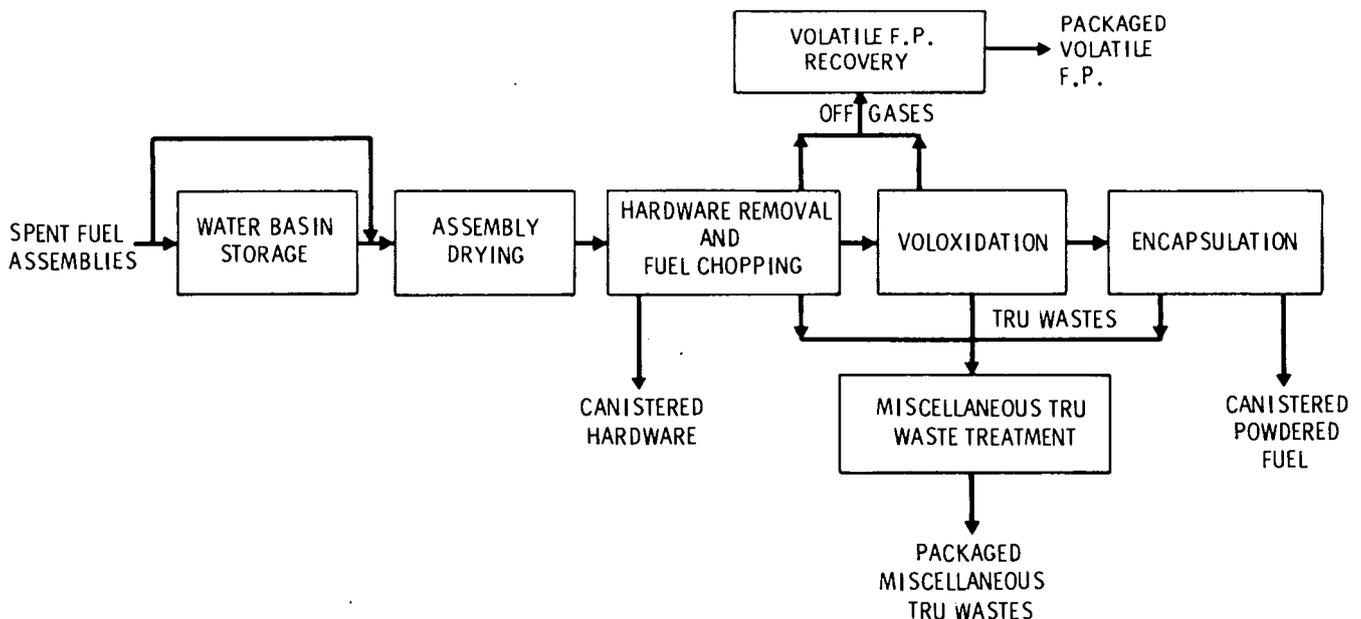


FIGURE 4.3.2. Flow Diagram for Encapsulation of Chopped and Voloxidized Spent Fuel

Combustible wastes produced during the processing (secondary wastes) are converted to ashes in an incinerator, and the ashes are blended with fixation materials and placed into waste containers. Incineration is accomplished in a molten salt combustion unit followed by fixation of TRU ashes in aluminum silicate mineral (clay). Noncombustible secondary wastes are also blended with fixation materials and placed into waste containers. Large pieces of failed equipment are disassembled or cut into smaller pieces suitable for packaging. The wastes requiring remote handling are packaged in 0.5 x 3.0-m cylindrical canisters, and the wastes suitable for contact handling are packaged in 55-gallon drums. The estimated numbers of these secondary waste packages considered to be TRU wastes are 30 canisters/GWe-yr and 6.5 drums/GWe-yr.

4.3.1.3 Dissolve Fuel and Convert to Glass

A process for dissolution of fuel and conversion to glass is described in ONWI-39 (Appendix C). This process incorporates fuel chopping and dissolution followed by concentration and calcination of the resultant solution followed by vitrification (conversion to glass) of the calcine. Voloxidation of the chopped fuel is also included in the process, as described in Section 4.3.1.2. A flow diagram for this process is shown in Figure 4.3.3. Although glass is the waste form described in ONWI-39, other waste forms such as those discussed in Section 4.3.2 could also be employed.

The voloxidized fuel is dissolved in nitric acid. During this operation the portions of the iodine and krypton that were not released to the off-gas system during voloxidation are evolved. The off-gas treatment process is described in Section 4.3.4.

The dissolution process also allows separation of the fuel cladding hulls from the fuel itself. The hulls are compacted in small containers with a hydraulic press and several of these containers are banded together and placed in a 0.5 x 3.0-m cylindrical canister. The required number of such canisters is estimated to be 17.5 per GWe-yr. The fuel assembly end fittings are packaged as described in Section 4.3.1.2.

The dissolved spent fuel is concentrated and then spray-calcined. The calcine is then fed along with glass frit into a continuous ceramic melter for vitrification. The molten glass that emerges from the melter is collected in canisters which, after cooling, are seal-welded. The referenced study uses 0.5 x 3.0-m cylindrical canisters; the number required is estimated to be 141 per GWe-yr. The number of canisters will vary however, depending on the thermal limitations of the final repository.

The miscellaneous combustible and noncombustible wastes and the failed equipment are treated the same as in the process described in Section 4.3.1.2. The estimated numbers of the TRU waste packages in this process are 43 canisters/GWe-yr and 9.4 drums/GWe-yr.

4.3.1.4 Dissolve Fuel for Disposal as a Liquid

The spent fuel treatment and packaging operations described in the preceding three sections result in waste packages suitable for geologic disposal. These operations could doubtless be adapted to provide different packages (if required) for disposal by some of the

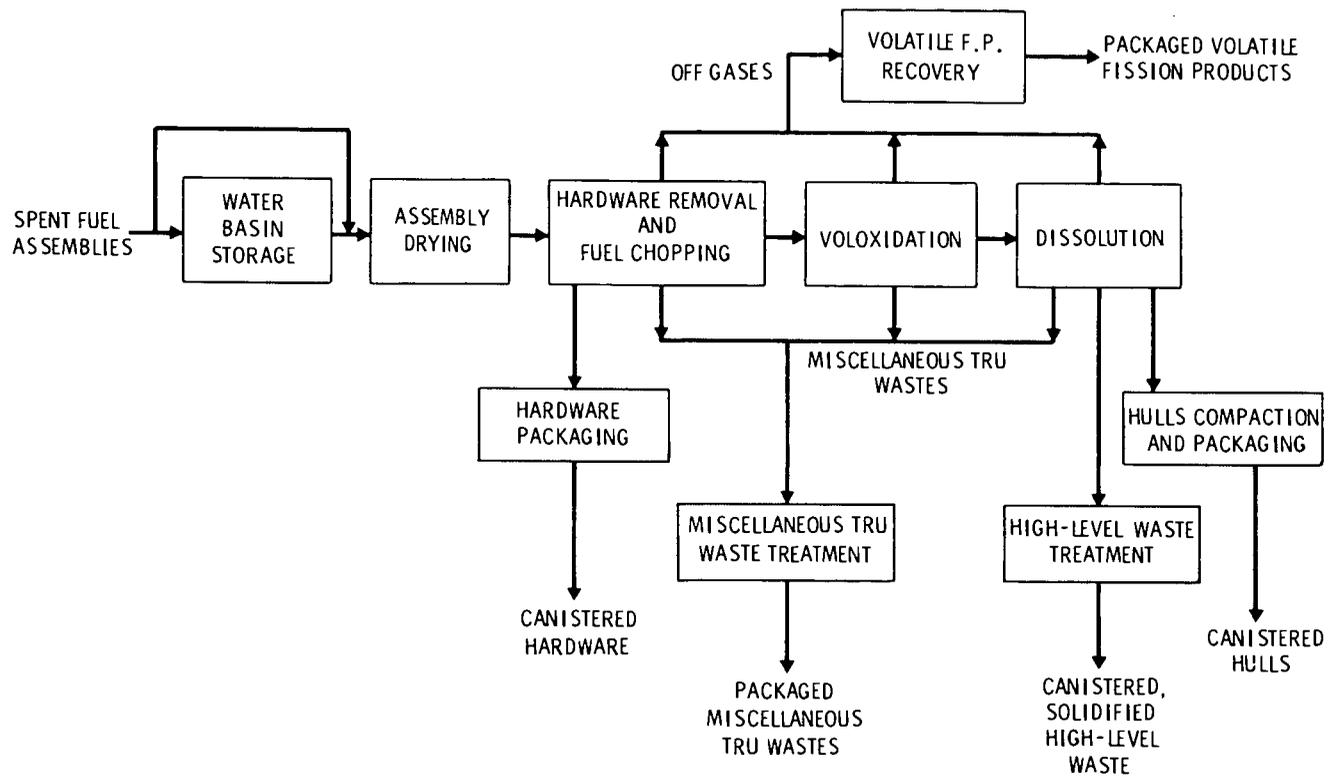


FIGURE 4.3.3. Flow Diagram for Encapsulation of Dissolved and Vitrified Spent Fuel

methods described in Chapter 6 as alternatives to geologic disposal. However, two of these alternative disposal methods (rock melting and well injection) involve disposal of the high-level waste in liquid form; thus, a modified spent fuel treatment process is required. Application of these methods to disposal of dissolved spent fuel presents added nuclear criticality safety problems and feasibility uncertainties resulting from the presence in the solution of all of the plutonium and the uranium.

By eliminating the calcination and vitrification operations, the spent fuel treatment process described in Section 4.3.1.3 could provide a liquid waste stream for disposal. Additional storage would probably have to be provided for the dissolved spent fuel solution to allow proper operation of the disposal process, however. A flow diagram for such a process is shown in Figure 4.3.4.

4.3.2 High-Level Liquid Waste Treatment

High-level liquid wastes are defined as "those aqueous wastes resulting from the operation of the first cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuels" (10 CFR 50). These wastes contain over 99% of the nonvolatile fission products and actinides, except U and Pu. If spent fuel is reprocessed, the U and Pu will normally be recycled. Only a small amount of U and Pu, perhaps 0.5%, resulting from waste losses during reprocessing will be in the HLW. Liquid high-level waste can be stored in tanks as an interim measure, but it must be solidified before transportation and disposal.

Many HLW treatment processes are under development and DOE is committed to examining the relative merits of many of these processes. For this discussion the candidate processes have been divided into three categories: those that convert the HLW into glass (Section 4.3.2.2), into a crystalline solid (Section 4.3.2.3), or into a composite or multiphase solid form (Section 4.3.2.4). A further important distinction concerning the candidate HLW waste treatment processes should also be made. The processes fall into two broad classes: those that have been developed to the stage of practical engineering-scale implementation, and those for which there has been some characterization of waste form properties but little or no process development. Calcine, low-melting glass and cement can be placed in the first category. All of the rest of the waste forms to be described fall into the latter, relatively undeveloped category. Additional data on many of these processes may be found in ERDA-76-43.

The processing descriptions given here assume that the HLW is not partitioned before treatment; however, because chemical partitioning has potential as a pretreatment for high-level liquid waste, partitioning techniques are also discussed in this section.

Before proceeding with the more general discussion, brief descriptions will be given of the two well developed high-level liquid waste treatment processes used in this Statement for evaluation of environmental impacts and costs. These processes are:

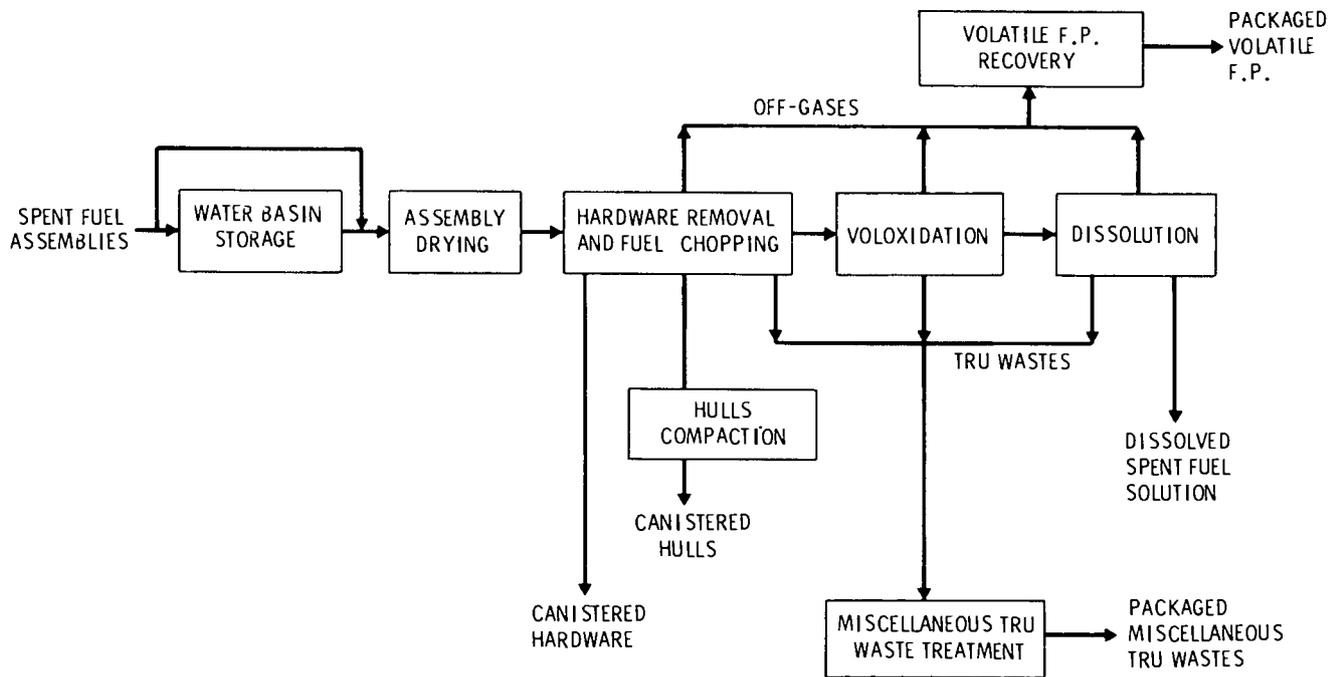


FIGURE 4.3.4. Flow Diagram for Dissolution of Spent Fuel for Disposal as a Liquid

1) vitrification by in-can melting following spray calcination and 2) fluidized bed calcination. These processes are described in detail in DOE/ET-0028. They produce a borosilicate glass product and a granular powder product, respectively.

Spray Calciner/In-Can Melting (Example Method)

A flow diagram for the in-can melting process, the example high-level waste solidification process of this Statement, is shown in Figure 4.3.5. The liquid HLW is dried and calcined in a spray-calciner, the resultant calcine is mixed with about twice its weight of glass-forming materials, and the mixture is melted within a steel canister. The filled canister is cooled and sealed by welding. The output of the example process amounts to about 2.2 m^3 of waste glass per GWe-yr; higher volumes would result from lower waste loadings. The number of canisters used to contain this volume of glass depends on a number of factors, among which are the heat generation rate of the contained waste and the heat generation rate per canister allowed by disposal considerations. For canister heat loadings of 1.2 to 3.2 kW (typical of those allowed in geologic repositories) and 6.5-year aged (out-of-reactor) waste, the number of canisters would amount to 44 and 17, respectively, per GWe-yr. A large variety of other glass-making processes have been developed; the output of these processes would be similar to that described here.

Fluidized Bed Calcination

In the fluidized bed calcination process (other calcination processes are also feasible), the liquid HLW is atomized as it enters the calciner vessel, which is heated by an in-bed combustion system. When the atomized HLW is injected into the hot bed, the waste constituents are converted to solids (primarily oxides) that adhere to the surface of particles already in the bed. The bed is fluidized by heated air entering through perforations in the bottom support plate. Calcined product is removed continuously so that the bed

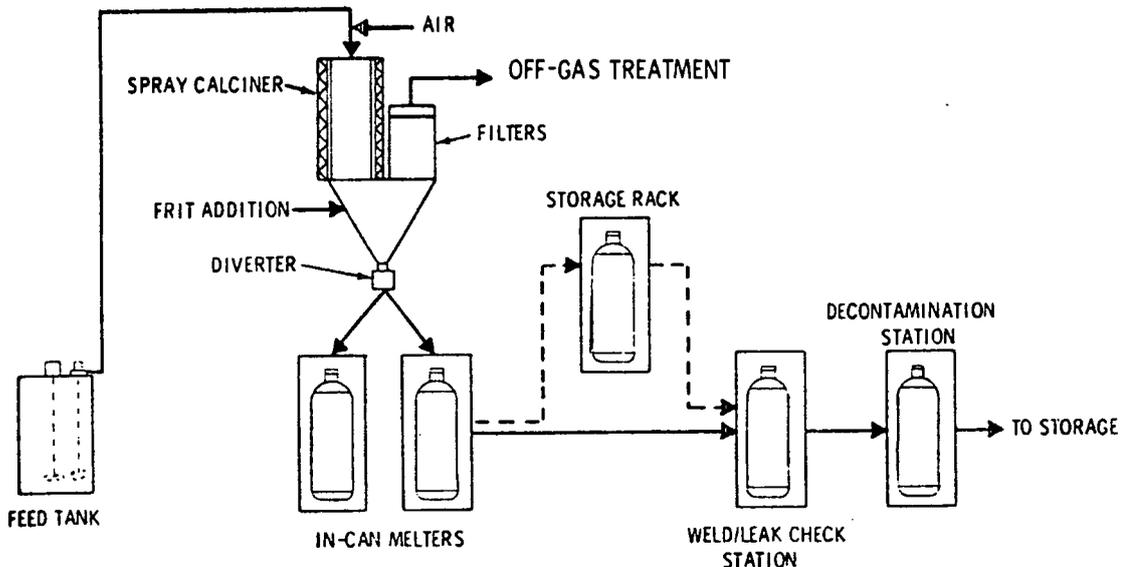


FIGURE 4.3.5. Flow Diagram for Spray Calciner/In-Can Melting Process

inventory remains essentially constant. The calcine is collected in canisters and residual water and nitrate are removed by heating to 700°C before the canisters are sealed shut. The output of the example process amounts to about 0.9 m³ of calcine per GWe-yr. A smaller diameter canister may be required for waste calcine than for waste glass to prevent overheating at the centerline of the canister, because of the lower thermal conductivity of calcine.

4.3.2.1 Chemical Partitioning

The partitioning or separation of certain elements from nuclear fuel cycle wastes has been viewed as a potential means for improving waste management (ERDA 1976, Campbell 1976, Schneider and Platt 1974, Cooperstein 1976). The perceived benefits result from removal of certain radionuclides and, hence, improvements in the management of the resulting partitioned radionuclide fraction compared to the management options for the unpartitioned wastes. Three subsequent options for disposal of partitioned radionuclides are discussed in this document: 1) transmutation as discussed in Section 6.1.7, 2) chemical resynthesis as discussed in Section 4.3.2.3, and 3) space disposal as discussed in Section 6.1.8.

In general, to partition simply means to separate elements, or groups of elements, from some mixture of chemical species. In a nuclear fuel cycle, partitioning would occur mainly during the reprocessing of spent fuel (ERDA 1976). There are many chemical elements in spent nuclear fuels (see Section 2.1), and many combinations in which these elements may be chemically separated. Consequently, there are also numerous partitioning alternatives that may facilitate useful waste treatment alternatives or disposal options. For all the specific partitioning candidates described here, one must realize that: 1) no partitioning processes have been demonstrated for waste disposal on a commercial scale; 2) historically most recovery processes leave several percent, or more, of the desired elements in the waste streams; and 3) partitioning for waste management purposes requires substantially higher recoveries than have been achieved to date. Partitioning itself is not an option for final disposal of radioactive wastes, although some waste partitioning may be required as a pretreatment to permit the final disposal of the resulting waste fraction (e.g., the partitioning of fission product iodine for space disposal).

With respect to waste management, partitioning may lead to improved waste characteristics for either the short term (less than 1000 years) or the long term (greater than 1000 years). The partitioning of strontium and cesium, for example, may be a useful option to reduce the self-heating (Buckingham 1967) characteristics of high-level wastes over the short term and thereby permit the storage of salt cakes that are not overly self-heating. In addition, the partitioning of actinides as well as some fission products may be useful to reduce the long-term radiotoxicity of wastes (Bond and Leuze 1975, Croff et al. 1977) and, therefore, reduce the exposure of future populations to radioactivity should the wastes ever be reintroduced into man's environment in the distant future (say after 100,000 years of storage).

Some partitioning options may be useful for maximizing energy conservation in the fuel cycle, facilitating the beneficial use (Rohrman 1968) of selected fission products, and improving nuclear safeguards (Campbell and Gift 1978; Pobereskin, Kok and Madia 1977). The recovery of cesium, for example, has been examined for use in sterilizing sewage sludges (Sivinski 1975; Reynolds, Hagengruber and Zuppero 1974); strontium might also be used as a heat source (Dix 1975) in remote and inaccessible areas. Partitioned palladium, rhodium, ruthenium and technetium could become mineral resources.

On the other hand, partitioning will invariably complicate waste management during the operation of the fuel cycle, as compared with other existing methods of dealing with the unpartitioned wastes (ERDA-76-43, Section 16.2). Several reasons for this are:

- Increased production of secondary wastes. Although the chemistry associated with the partitioning of radionuclides is quite diverse, all known options generate significant quantities of secondary wastes that must be managed. These secondary wastes may be treated by incineration, by compaction, by immobilization, or by other methods, but invariably the waste volumes will be increased by the partitioning, and waste management costs will also increase. Many partitioning options will significantly increase the high-level waste volume because of the addition of salting agents or other nonvolatile species. Also, many chemical additives may adversely affect high-level waste solidification and the long-term stability of the waste form (e.g., glass devitrification).
- Increased transportation costs and requirements. Most partitioned waste fractions can be transported safely only with extensive shielding. For many of the transmutation cycles, the transmutable elements are recycled many times before a significant reduction in quantity is achieved. In the case of actinides some of the transmuted products are strong neutron emitters and will constitute a handling problem.
- Increased costs due to partitioning and secondary waste treatment. All known partitioning options involve sophisticated chemical separation processes that must be remotely maintained and operated. Significant capital investment and operating costs will result if these chemical processes are implemented. The recovery of selected waste constituents, like cesium and strontium, does not significantly reduce the cost of managing the residual high-level waste.
- Increased potential for radiation exposure. Since partitioning will require increased chemical operations, handling, transportation and storage, the potential for increased occupational radiation exposure also exists. The potential for accidental release of radioactive material (and general population exposure) will also be increased. These factors must be quantified if partitioning is adopted.
- Increased thermal loading. Partitioned waste fractions with high heat generation densities impose a higher thermal load on containment materials than does unpartitioned waste. A recent study (NAS 1978) has suggested that the permanent containment of cesium and strontium partitioned from wastes at Hanford will be difficult because of the high heat densities involved.

4.3.2.2 Glass Waste Forms

Vitrification (conversion to glass) of high-level liquid wastes is being developed in Germany, France, India, Russia, Great Britain, Belgium, Japan, Canada and the United States. A facility for vitrification of the HLW from the Marcoule reprocessing plant has been operating in France since the summer of 1978 (Bonniand et al. 1978). The various HLW vitrification processes and properties of the glasses made by them have been well described in recent reports and symposia proceedings (McCarthy 1979, Chikalla and Mendel 1979).

Low-Melting Glasses

Low-melting glasses are glasses that can be processed at temperatures below about 1200°C. The most well developed vitrification processes throughout the world all produce low-melting glasses of a borosilicate formulaion, although a small amount of development continues on phosphate glass formulations (Kelley 1975, Wiley and LeRoy 1979, Gombert et al. 1979, Kupfer 1979 and Mendel 1978). The product of these borosilicate glass processes is a glass casting in a metal canister. The castings vary in size depending on the process and the amount of radioactivity, but are generally cylinders from 0.3 to 0.6 m in diameter and 1 to 3 m long.

Borosilicate waste glasses can contain one-third or more (by weight) HLW oxides; the remainder is inert glass-forming material added during vitrification processing. The glasses can tolerate wide variations in HLW composition without sacrificing their properties. The glass castings contain some fractures caused by thermal stresses induced as the large monoliths cool. Waste glasses are metastable materials and they must be cooled fairly rapidly (a cooling rate of at least 10°C/hr between 900°C and 600°C is satisfactory for most formulations) to prevent excessive devitrification from occurring. At lower temperatures, e.g., those encountered in geologic disposal, the rates of thermal devitrification are too slow to be a factor. Extensive studies have shown that the only significant effect of devitrification, if it does occur, is a small increase in leach rate. The increase is usually less than a factor of three even in fully devitrified glasses but in some formulations may be as high as 10. The glass phase exhibits excellent stability in radiation fields as shown by tests simulating over 500,000 years of alpha radiation.

Borosilicate waste glasses also exhibit good chemical durability; however, there is a finite reaction rate with water. The reaction rate is dependent on many factors but for typical waste glasses is usually in the range 10^{-7} to 10^{-5} g glass/cm²-day after a few weeks of leaching at 25°C. The rate increases with temperature, rising a factor of 10 to 100 for a 100°C increase in temperature.

High-Temperature Glasses

In the context of this discussion, these are glasses that melt above 1200°C. They contain more silica or alumina than the low-temperature glasses. An early example of a high-temperature waste glass is the nepheline syenite waste glass made in Canada from 1958 to 1960. Blocks of this glass, without canisters, were buried below the water table at Chalk River in 1960. The leaching behavior of these glass blocks has been monitored by means of

wells. The ^{90}Sr leach rate decreased with time and after about 5 years stabilized at the very low rate of 5×10^{-11} g glass/cm²-day (Merritt 1977).

Recently, development of a stuffed glass process has begun at Catholic University in Washington, D.C. (Simmons et al. 1979). The process utilizes a high-temperature, high-silica glass that can be prepared in a porous form outside the radioactive processing cell. The pre-prepared porous glass is then soaked in HLW solution. After a suitable soaking period the solution-laden porous glass is removed from solution and the HLW constituents are precipitated. The porous glass is then soaked in a solvent that removes the waste from a surface layer of the porous glass. The solvent is subsequently evaporated and the porous glass is dried at 625° to 700°C to convert the HLW constituents in the pores to oxides. Then the temperature is raised to 900°C for sintering. During sintering, the pores collapse. The final product is solid glass that contains the radioactive waste materials interstitially, and has a high-silica envelope on the outer surface. Alternatively, the same final form can be obtained by putting waste-laden porous glass granules in an envelope of waste-free porous glass and sintering to close the pores.

The stuffed glass process potentially yields a product with the durability of a high-melting glass but utilizes lower processing temperatures. In addition, the product has a built-in barrier of inert high silica glass on the surface.

Glass-Ceramics

Glass-ceramics are a class of specially formulated materials that can be melted, processed and formed as glasses and then devitrified, or crystallized, under controlled conditions. Glass-ceramics have become important commercially in the last 20 years. They are valued for their thermal stability and physical ruggedness.

Most of the investigations of glass-ceramics as materials for HLW disposal have been carried out in Germany at the Hahn-Meitner Institute in Berlin and at Karlsruhe (De et al. 1976, Guber et al. 1979). The waste-containing glass-ceramics formulated to date are usually only about 50% crystalline (commercial glass-ceramics are over 95% crystalline). Some improvements in thermal stability (higher softening points) and physical ruggedness have been observed; the leach rates obtained to date are in the same range as those of low-melting waste glasses.

4.3.2.3 Crystalline Waste Forms

For the purposes of this discussion all nonvitreous high-level solid waste forms will be termed crystalline. In general, crystalline waste forms, particularly those that have undergone extensive thermal treatment and are not approaching solid solution limits, are thermodynamically more stable than glass waste forms. In some crystalline waste forms the crystals are "tailored" to resemble minerals that have a demonstrated stability in nature.

Cement

Cements are used routinely to encapsulate low- and intermediate-level radioactive wastes. Liquid or slurry wastes are mixed with a predetermined weight of dry solids. The

solids may be primarily Portland cement such as used in concrete, or may consist of cement mixed with fly ash and clays (grouts) and can be specially designed (usually high alumina) cements (Stone 1977 and Lokken 1978).

Cements are intrinsically somewhat porous and due to the hydrated phases are potentially sensitive to damage from radiation and long-term thermal exposure. They have been considered for the treatment of defense HLW, and techniques that reduce the porosity and water content may even make their use for commercial HLW feasible (Roy and Gouda 1978). One such technique is the FUETAP process being investigated at Oak Ridge National Laboratory in which the waste-containing cement mixture is processed at 250°C and 600 psi (Moore et al. 1979).

Calcine

Defense HLW has been calcined using a fluidized bed calcination process at the Idaho Chemical Processing Plant (ICPP) since 1963. Over 1500 m³ of granular calcined waste particles are now stored in stainless steel bins housed in underground concrete vaults. The calcined waste is a good low-volume, noncorrosive form for storage.

The ICPP calcination process converts the HLW to dry salts and oxides. Consolidation techniques that decrease the surface area of the solids, decrease the potential for airborne fines, and increase the chemical durability are being investigated. The consolidation techniques are either sintering processes that yield a type of glass-ceramic product or processes that embed the pelletized calcine in an inert matrix (INEL 1978, Lamb et al. 1979, see Section 4.3.2.4).

Synthetic Minerals

To create synthetic minerals, nuclear waste constituents are chemically incorporated in crystalline mineral species. The long-term stability of synthetic mineral waste forms can be deduced from the known behavior of analogous naturally occurring minerals. Of course, unavoidable differences, such as radiation effects, must be studied. A review of the stability of minerals that could contain radionuclides is given in Appendix P of Volume 2.

Development of one synthetic mineral concept (called supercalcine) began at Pennsylvania State University and the Pacific Northwest Laboratory (McCarthy 1977, 1979a; McCarthy and Davidson 1975). The concept may be considered an evolution of the well-developed calcination processes. Instead of calcining the liquid HLW as received, additions of calculated quantities of Ca, Al, Si, etc. are made to the HLW so that after calcination and a heat treatment the waste constituents are chemically bound in predetermined mineral assemblages. However, because HLW contains so many different elements, the mineral assemblages tend to be very complex and difficult to characterize. Recently the emphasis in some investigations has switched to the development of stable synthetic minerals for only the actinides in the waste. Fluorite and monazite structures appear to form very stable crystals containing these long-lived waste constituents (McCarthy 1979b). Hot pressing techniques are being investigated for consolidation of the synthetic mineral calcines.

Another synthetic mineral concept being studied extensively is Synroc, an acronym for synthetic rock coined by Dr. A. E. Ringwood of the Australian National University at Canberra, for a concept in which the radionuclides are incorporated in solid solution in just three nonsilicate minerals: hollandite, perovskite and zirconolite (Ringwood et al. 1979). A distinguishing feature of this concept is that it maintains a low waste loading (≤ 10 wt%) so that the known stability of the host crystals is not perturbed. The waste forms are made by mixing calcined HLW with the Synroc additives and hot pressing at 1200° to 1300°C in sealed nickel containers.

One method of obtaining good accommodation of waste radionuclides in synthetic mineral assemblages is to limit the waste loading, as the Synroc concept does. Conceptually, partitioning the HLW into fractions would simplify the task even further and could permit a higher waste loading. The waste would be partitioned based on considerations of chemical and mineralogical similarities, and the availability of techniques for isolating various waste fractions. The possibility exists of processing each fraction individually into a different synthetic mineral. This concept minimizes crystal compatibility problems during processing and opens up the possibility of using multiple repository sites selected for stability with the various synthetic mineral assemblages made from each fraction.

4.3.2.4 Composite Waste Forms

In composite waste forms, the HLW is usually contained in particles or spheres of one type of material, which is surrounded by one or more different nonradioactive materials. The materials are chosen to have properties that complement one another, so that the properties of the composite are superior to those of the HLW-containing material by itself. The waste-containing material can be particles, spheres, or small pieces of any of the candidate waste forms described in Sections 4.3.2.2 and 4.3.2.3; the surrounding materials are metals or ceramics used to increase thermal conductivity and/or fracture resistance, and possibly to act as additional barriers to the release of radionuclides from the waste-containing core material.

Metal Matrices

The use of metal matrices in composite waste forms has been studied for many years (Lamb 1979, Jardine and Steindler 1978, Neumann 1979). Metal matrices are used to improve thermal conductivity and to minimize fracturing of the waste glass beads by adding ductility, i.e., an ability to bend without breaking, to the composite waste form. A radioactive demonstration of the PAMELA process, in which HLW glass beads are embedded in a lead matrix, is planned as a joint German-Belgium project in the early 1980s (Salander and Zuhlke 1979).

Low-melting metals, such as lead or aluminum and their alloys, have received the most consideration as waste form matrices, but higher-melting metals, such as copper and even steel, can be used to form porous matrices by a powder sintering technique. Even nonporous melt-formed metal matrices may not form a complete barrier to leaching if water contacts the waste form; the bond between the metal and the waste-containing particles may not be tight

enough to prevent access of water to the interior of the composite. A barrier can be formed, however, as is done in the PAMELA process, by suspending the waste-containing particles in a basket in the canister and filling the annulus between the basket and the canister wall with pure matrix metal.

Coated Particles

Coated particle composite waste forms are being developed, partially based on the technology developed for the manufacture of high temperature gas-cooled reactor (HTGR) fuels (Rusin et al 1978, 1979a and 1979b). These fuels consist of ceramic pellets that are coated with pyrolytic graphite and silicon carbide, and embedded in a graphite matrix. The core material that has been most studied for coated particle composite waste forms is the synthetic mineral calcine described in Section 4.3.2.3; however, the concept can utilize other core materials. Calcine pellets are formed in a disk pelletizer and coated with pyrolytic graphite and silicon carbide in a fluidized bed. Laboratory tests have shown that an outer coating of durable Al_2O_3 can be added. The coated particles would be surrounded by a metal matrix in canisters before emplacement in a geologic repository.

Coated particles are a way of adapting the multiple barrier concept to the waste form itself. Tests have shown that the particles can have very good chemical durability. However, the processing would be very complex and require a large amount of development before it could be done remotely.

Cermet

This waste form concept, under development at Oak Ridge National Laboratory, produces a uniform dispersion of waste oxide particles within a metal matrix (Quinby 1978). The waste and specific additives required to form the desired ceramic oxide phases and metal alloy matrix are dissolved together in molten urea. The urea solution is precipitated and calcined and the fine powders produced in this step are compacted by extrusion or pressing into desired shapes. In the final processing step the reducible metal oxides, such as oxides of Cr, Ni, Fe, and Co, are reduced in a H_2 or CO atmosphere to form an alloy that encapsulates the unreduced ceramic oxides. After the $800^\circ C$ reduction the composite is mixed with an organic binder, extruded to form rods and sintered in a nonoxidizing atmosphere at $1200^\circ C$ to form a dense compact.

High waste loading can be achieved in cermets because metals from salts present in the waste form part of the metal matrix. The reducing conditions reduce volatilization problems during processing.

4.3.2.5 Waste Form Characterization

In that the DOE is committed to examining the relative merits of all potentially available waste forms, research and development is being supported on almost all of the waste forms described in Sections 4.3.2.2, 4.3.2.3 and 4.3.2.4. Treatment processes are already available to produce certain of the waste forms, such as low-melting glass. The DOE program is designed to determine if there are other waste forms that can be practically produced and that offer improved characteristics. A Materials Characterization

Center has been set up to provide techniques for comparing important waste form materials characteristics on a common basis (Nelson et al. 1980). The first issue of the Nuclear Waste Materials Handbook will be published in approximately two years. It will contain materials data, not only for candidate waste forms, but also for other waste package components.

Since the most likely mechanism for release of radionuclides to the biosphere is reaction with and transport by ground water, resistance to leaching of radionuclides by ground water is the performance characteristic of major interest. Leach resistance can be highly dependent upon the physical, chemical, mechanical, and radiation stability of the waste form. The stability of a waste form depends upon its response to radiation, temperature, and the chemical environment (Mendel et al. 1975). The factors influencing long-term stability are: 1) transmutation by radioactive decay, which may alter the chemical structure of the waste form; 2) recoil from alpha decay, which may break chemical bonds and alter the physical structure of the waste form; 3) heat generated by radioactive decay, which may cause the waste form to change to a more thermodynamically stable state and which may accelerate potential chemical reactions, including leaching; and 4) the chemical environment, i.e., water plus dissolved ions, which ultimately determines the rate of release of radioactive materials into the repository.

4.3.3 TRU Waste Treatment in the Reprocessing Cycle

When spent fuel is reprocessed for uranium and plutonium recycle, the non-high-level and nongaseous wastes that result from these operations and from the mixed oxide fuel fabrication must also be treated and packaged. This section addresses the treatment of these solid and liquid TRU wastes. Treatment and packaging processes for such wastes are described in detail in DOE-ET-0028 (Section 4.0), where wastes are discussed in four categories: 1) fuel residue (the fuel hulls and assembly hardware), 2) failed equipment and noncombustible waste, 3) compactable and combustible waste, and 4) wet and particulate solid wastes. Brief descriptions of the treatment processes for these wastes are given in the following sections; the referenced document may be consulted for details. Both TRU and non-TRU wastes of the latter three categories result from operation of fuel reprocessing plants (FRPs). Only the treatment of the TRU wastes is considered in this Statement; the treatment of the non-TRU portions would be similar, however.

4.3.3.1 Fuel Residue Treatment

Packaging without compaction is the example fuel residue treatment process used in this Statement. Mechanical compaction of hulls and melting of hulls are also described to illustrate other alternatives. The fuel residue packages have surface dose rates well above 0.2 R/hr. Remote handling of these wastes is thus required.

Fuel Residue Packaging Without Compaction (Example Method)

Packaging without compaction is a treatment concept in which the nonsegregated fuel residue is monitored for undissolved fuel, dried, and sealed without compaction in stainless

steel canisters (0.76 m dia x 3 m) for shipment to interim storage or to a repository. The void spaces in the canister are filled with dry sand to reduce the possibility of ignition of Zircaloy fines in the fuel residue. Alternatives within the packaging without compaction concept involve separate packaging of the hulls and hardware, deactivation of fines before packaging, and use of filler materials other than sand (e.g., concrete). Other containers (e.g., 55-gallon drums) could also be employed.

Figure 4.3.6, the flow diagram for fuel residue packaging without compaction, shows the steps involved in the process. The quantity of packaged waste resulting from this option is estimated to be 9.1 canisters/GWe-yr.

Mechanical Compaction of Hulls. Mechanical compaction of hulls is a treatment concept for fuel residues in which the hulls are separated from the fuel assembly hardware and Zircaloy fines, compacted to 50% of theoretical density, and packaged in stainless steel canisters (0.76 m dia x 3 m) for shipment to interim storage or to a repository. The Zircaloy fines are deactivated by oxidation and packaged in identical canisters along with the fuel assembly hardware. Compaction of the hulls could be done by a variety of processes, none of which has been evaluated with irradiated hulls. Hydraulic press compaction was selected as the alternative most technically feasible at present.

The steps of the compaction packaging concept are shown in Figure 4.3.7. Implementation of this option is estimated to result in 1.6 canisters/GWe-yr of fuel hardware and 3.8 canisters/GWe-yr of compacted hulls.

Hulls Melting Process

The hulls melting concept considered here uses the Inductoslag melting process developed by the U.S. Bureau of Mines Metallurgical Research Center in Albany, Oregon. In this

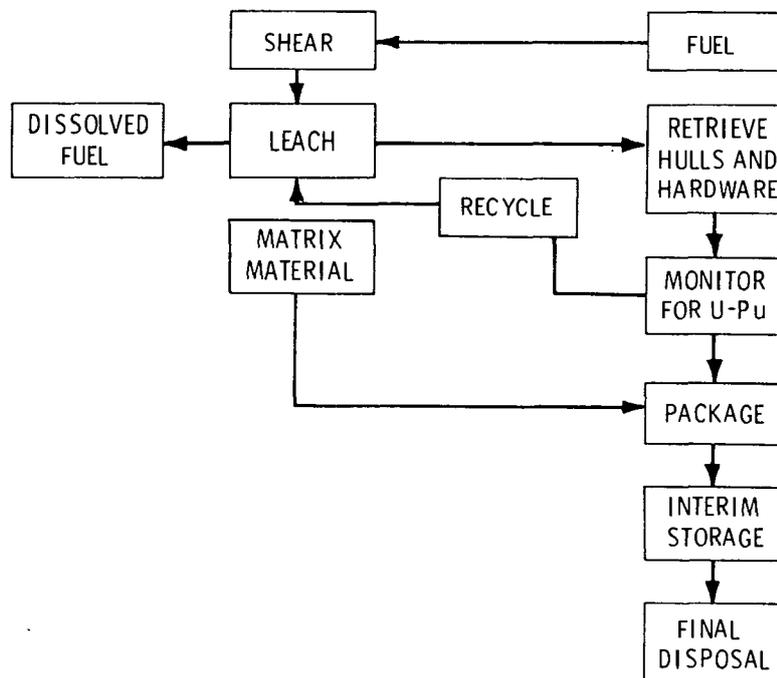


FIGURE 4.3.6. Flow Diagram for Fuel Residue Packaging Without Compaction

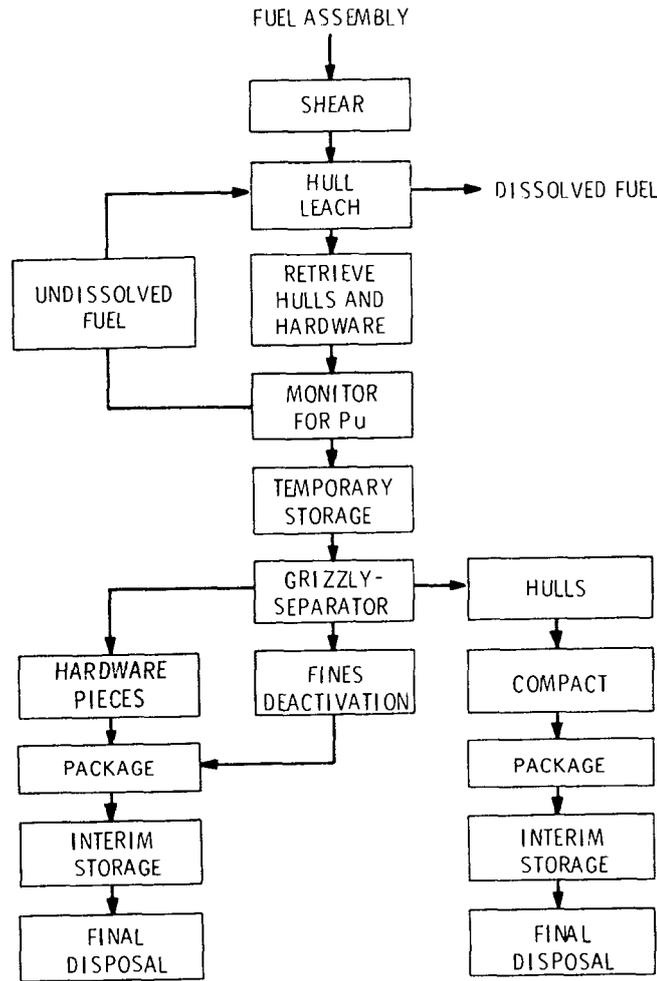


FIGURE 4.3.7. Flow Diagram of Mechanical Compaction of Hulls

process, the sheared cladding hulls are segregated from the stainless steel end fittings and other fuel element hardware and from the Zircaloy fines. The hulls are melted, and the ingots from the melter are sealed into stainless steel containers. The Zircaloy fines are deactivated to eliminate pyrophoric hazards and are packaged with the stainless steel components without melting. This melting concept has been demonstrated successfully in making ingots 10 cm (4 in.) in diameter from simulated fuel residue.

A flow diagram for the melting process is identical to that shown in Figure 4.3.7 except that melting is substituted for compaction. The facility is designed to produce 6 ingots/day, 0.23 m dia x 1.45 m long. These ingots are packaged in 0.76 m dia x 3 m stainless steel canisters, and the fuel hardware is packaged in identical canisters. The estimated quantities are 1.6 canisters/GWe-yr of hardware and 2.1 canisters/GWe-yr of melted hulls.

4.3.3.2 Failed Equipment and Other Noncombustible Waste Treatment

The example treatment of failed equipment and noncombustible waste used in this Statement involves decontamination and disassembly of some of the failed equipment (but not of

noncombustible waste), and packaging either in 55-gallon drums, in 1.2 x 1.8 x 1.8 m steel boxes, or (at an FRP) in canisters like those used to contain fuel residue (Section 4.3.3.1). Failed equipment is packaged in canisters when it cannot be decontaminated sufficiently to allow packaging in boxes (the boxes must have a surface dose rate less than 200 mR/hr) or it cannot be disassembled to fit in drums. Figure 4.3.8 is a schematic flow diagram illustrating treatment procedures at an FRP. Procedures at a MOX-FFP are similar in most respects. Alternative treatment concepts involve varying degrees of decontamination and disassembly before packaging and the addition of fixation materials (e.g., cement) within the packages.

For the generic reprocessing cycle studied (Section 3.2.1.2), it is estimated that the quantity of failed equipment resulting from operation of an FRP could be contained in a mixture of packages comprising 1.4 canisters/GWe-yr, 1.1 boxes/GWe-yr, and 9.0 drums/GWe-yr. The boxes have surface dose rates low enough to allow contact-handling but the canisters and drums require remote handling. The noncombustible waste is packaged only in 55-gallon drums; the estimated quantity from an FRP is 84 drums/GWe-yr, approximately 10% of which

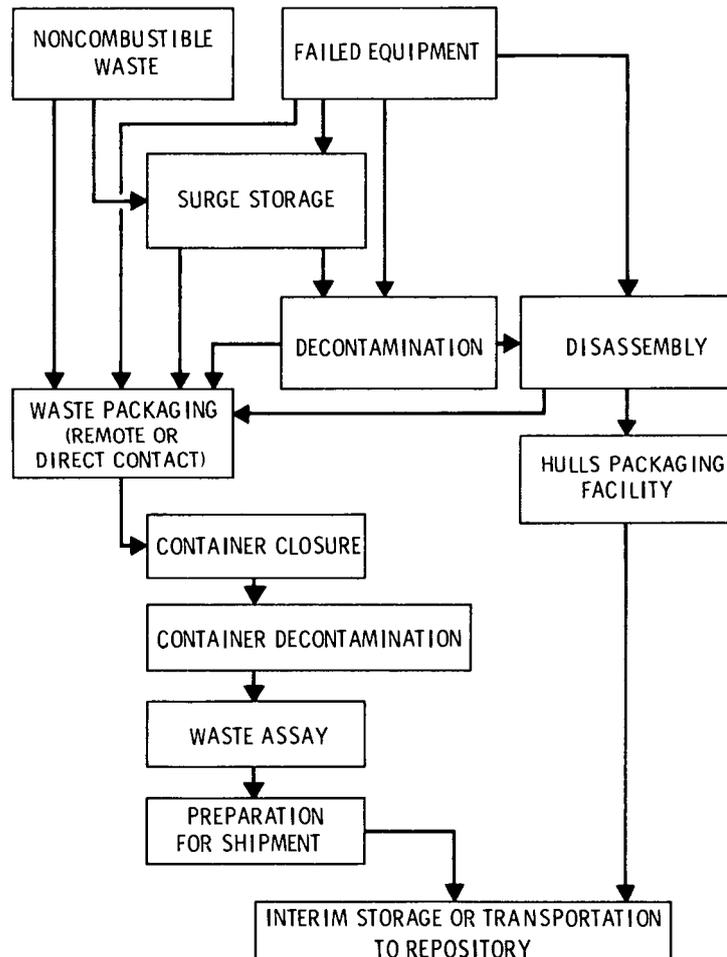


FIGURE 4.3.8. Flow Diagram for Treatment of Failed Equipment and Noncombustible Waste at an FRP

may be contact-handled. The quantities of failed equipment and noncombustible wastes estimated for a MOX-FFP could be contained in a mixture of packages comprising 0.38 boxes/GWe-yr and 7.5 drums/GWe-yr. All of these packages could be contact-handled.

4.3.3.3 Combustible and Compactable Waste Treatment

Three major alternatives have been used for treating general trash and combustible waste: incineration, packaging without treatment, and compaction. Incineration consists of burning the waste and treating the off-gas for removal of radionuclides and other noxious materials, thereby decreasing the waste volume and rendering it noncombustible. Incineration also reduces the potential of biological action occurring in the waste. Packaging without treatment consists of simply packaging general trash and ventilation filters in steel drums for interim storage or interment at the repository. The third alternative, compaction, consists of compacting the waste and packaging it in steel drums for interim storage or interment at the repository. All three methods have been widely used in the nuclear industry, although incineration has not been applied to wastes requiring remote handling. The latter two methods may not give waste packages that meet waste package criteria for the repository.

Incineration was chosen as the example treatment process for this Statement because it both renders the waste noncombustible and reduces the volume. Several incineration processes have been successfully operated with radioactive combustible wastes (Perkins 1976, Borduin and Tobaos 1980). The process assumed here and described in DOE/ET-0028 employs a controlled-air, dual-chamber incinerator. Packaging without treatment was also examined in detail as an alternative since it represents the other end of the spectrum in terms of cost, volume reduction, and flammability of the packaged waste.

Incineration (Example Method)

The FRP wastes include both materials that must be handled remotely and those that can be contact-handled; we assume the use of separate but identical incinerators for the two waste categories. The wastes sent to these two units are sorted and high-density combustibles are shredded, as are wooden filter frames after filter media have been removed in a filter media removal and pelletizing press. Pelletized filter media and noncombustibles are packaged in 55-gallon drums for disposal. The sorted and shredded combustibles are incinerated, and the ash (which contains essentially all of the radionuclides present in the waste) is collected for transfer to the wet waste and particulate solids immobilization facility. The off gas from the incinerator is sent through a high-energy gas-scrubbing system for cooling and for removal of volatilized radionuclides, acidic gases, and particulates before being filtered and routed to the FRP atmospheric protection system. The scrubbing solution is concentrated and sent, along with the ash, to the wet waste and particulate solids immobilization facility. Figure 4.3.9 provides a simplified flow diagram of these operations.

We assume that the MOX-FFP is located apart from the FRP and that a separate incineration facility is therefore required. The facility design is nearly identical to that in the FRP; however, because of the relatively small volume of off-gas scrubbing solution, it does not provide for solution concentration before immobilization.

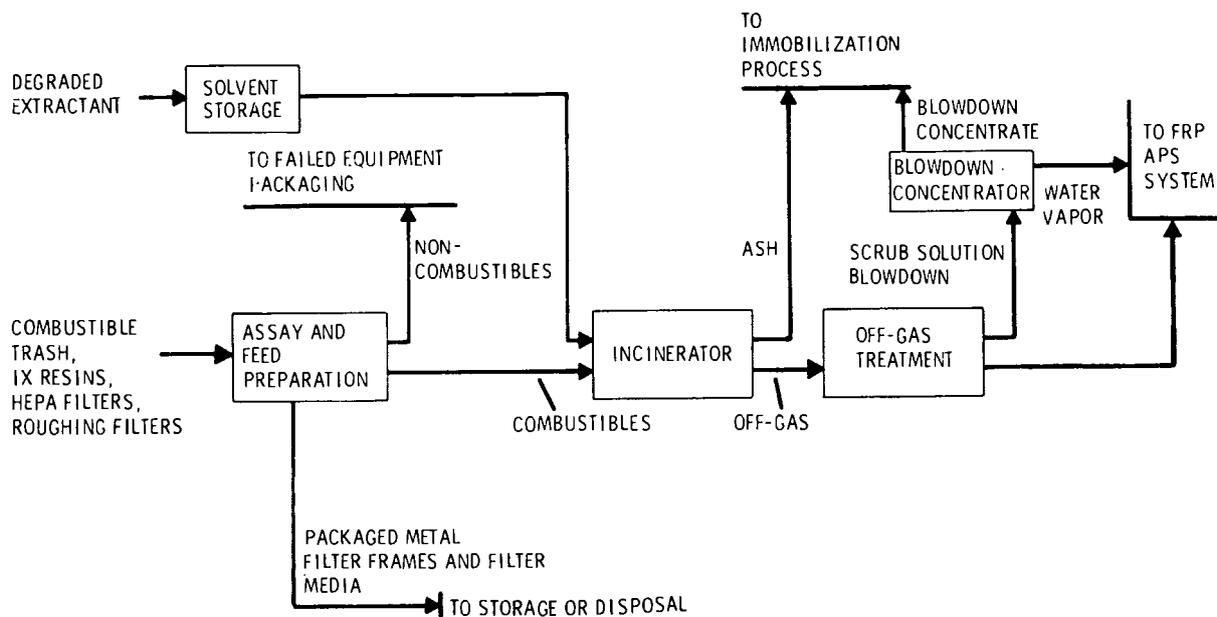


FIGURE 4.3.9. Treatment of Combustible Wastes and Filters at FRP Remotely Handled Waste Incinerator Facility

The only packaged waste outputs from the example incineration facilities are the drums containing the pelletized filter media and minor amounts of noncombustible waste and crushed metallic frames from HEPA filters. The estimated quantities would fill 7.6 55-gallon drums/GWe-yr from FRP operation and 0.95 55-gallon drums/GWe-yr from MOX-FFP operation. The drums from the FRP would require remote handling, but those from the MOX-FFP (because the principle activity results from alpha radiation) could be contact-handled.

Packaging Without Treatment

The waste packages employed for packaging combustible and compactable wastes without treatment are steel drums; the larger HEPA filters are packaged in 80-gallon drums, and the remaining wastes are packaged in 55-gallon drums. The wastes are assumed to be sealed in plastic bags before they are shipped to the packaging facility. In the packaging facility they are examined and placed in new drums (if necessary), assayed for fissile material content, and the lids are tightened to the drums.

The quantities of packaged waste are quite large under this option. We estimate 55 80-gallon drums/GWe-yr and 137 55-gallon drums/GWe-yr of remotely handled waste and 228 55-gallon drums/GWe-yr of contact-handled waste from the FRP. For the MOX-FFP the estimates are 6.6 80-gallon drums/GWe-yr, and 21.5 55-gallon drums/GWe-yr, all of which could be contact-handled.

If the packaging without treatment option is implemented, alternative treatments are employed for two types of combustible waste: ion exchange resins and degraded extractant. The ion exchange resins are sent to the wet waste and particulate solids immobilization facility, and the degraded extractant is burned in an incineration unit designed specifically for that purpose.

4.3.3.4 Immobilization of Wet Wastes and Particulate Solids

Prior to shipping and isolating wet wastes, they must be immobilized. This step may be done by a variety of methods. Immobilization of these wet wastes in bitumen and cement (bituminization and cementation) is discussed here as applied to an FRP and a MOX-FFP. Another alternative, urea-formaldehyde immobilization, requires process equipment similar to that for cementation. Cementation is the example treatment process chosen for this Statement.

Cementation (Example Method)

Immobilization of radioactive wet wastes in cement involves mixing the wastes with cement, placing the mixture into drums, and allowing the mixture to harden to a liquid-free product. Cement immobilization of radioactive wastes has been widely used in the U.S. A variety of cementation technologies have been developed, including in-drum mixers, drum tumblers, and in-line mixers, each of which is described in ERDA-76-43. For this Statement, a drum-tumbling system was selected for the following reasons:

- Both liquid and dry wastes can be immobilized without altering the commercially available technology.
- The wastes are mixed inside the drums, preventing external solidification of the waste-cement mixture.

The process flow diagram for a cementation system at an FRP is shown in Figure 4.3.10. A similar system can be used at a MOX-FFP after neutralization of the acidic liquids and treatment to remove the ammonia present in those wastes (to avoid possible later pressurization of sealed containers).

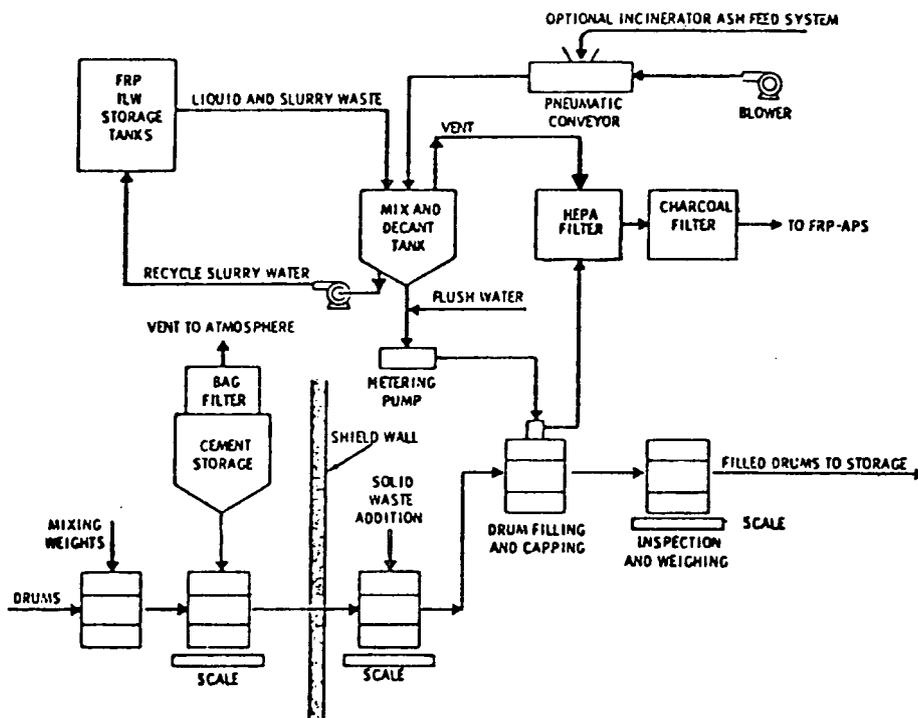


FIGURE 4.3.10. Process Flow Diagram for Cementation at Fuel Reprocessing Plant

The packaged waste output of the cementation systems depends markedly on whether or not the combustible wastes are incinerated (because the incinerator ash and scrubber solutions are additional feeds to the cementation systems). If the combustible wastes are incinerated, the output of the cementation systems will be 106 55-gallon drums/GWe-yr at an FRP and 31 55-gallon drums/GWe-yr at a MOX-FFP. About 40% of the drums originating at an FRP and all of the drums originating at a MOX-FFP could be contact-handled.

If the combustible wastes are not incinerated, the packaged waste output of the cementation systems will be 49 55-gallon drums/GWe-yr at an FRP and 11 55-gallon drums/GWe-yr at a MOX-FFP. All of the drums originating at an FRP require remote handling, but those originating at a MOX-FFP would be contact-handled.

Bitumenization

Immobilization of radioactive wet wastes in bitumen involves mixing the waste with liquid bitumen or asphalt binder and placing it in 55-gallon drums. The temperature of the binder at the time of mixing (above 100°C) evaporates the free water, and thus reduces the waste volume. Use of bitumen to immobilize radioactive wastes has been well demonstrated, largely through extensive operating experience in Europe. However, it is uncertain whether bitumenized waste forms will meet waste form criteria for repositories.

Several types of bitumenization processes have been developed as discussed in ERDA-76-43. In this Statement, a continuous screw extruder process was considered for the following reasons:

- The screw extruder bitumenization process operates at lower temperatures and with shorter residence times than the batch process, thus minimizing off-gas problems.
- The process uses well-demonstrated technology.
- The process is commercially available in the U.S.

A process flow diagram for a bitumenization system at an FRP is shown in Figure 4.3.11. A similar system can be used at a MOX-FFP after neutralization of acidic liquids.

If the combustible wastes are incinerated, the packaged waste output of the bitumenization systems will be 48 55-gallon drums/GWe-yr at an FRP and 10 55-gallon drums/GWe-yr at a MOX-FFP. About 2% of the drums originating at an FRP and all of the drums originating at a MOX-FFP could be contact-handled.

If the combustible wastes are not incinerated, the packaged waste output of the bitumenization systems will be 26 55-gallon drums/GWe-yr at an FRP and 8.7 55-gallon drums/GWe-yr at a MOX-FFP. About 3% of the drums originating at an FRP and all of those originating at a MOX-FFP could be contact-handled.

4.3.4 Gaseous and Airborne Waste Treatment

Spent nuclear fuel contains some radionuclides that are released in gaseous form during certain treatment operations. Such volatile radionuclides include the fission products ^3H , ^{85}Kr , and ^{129}I and the activation product ^{14}C . A small portion of the fission product

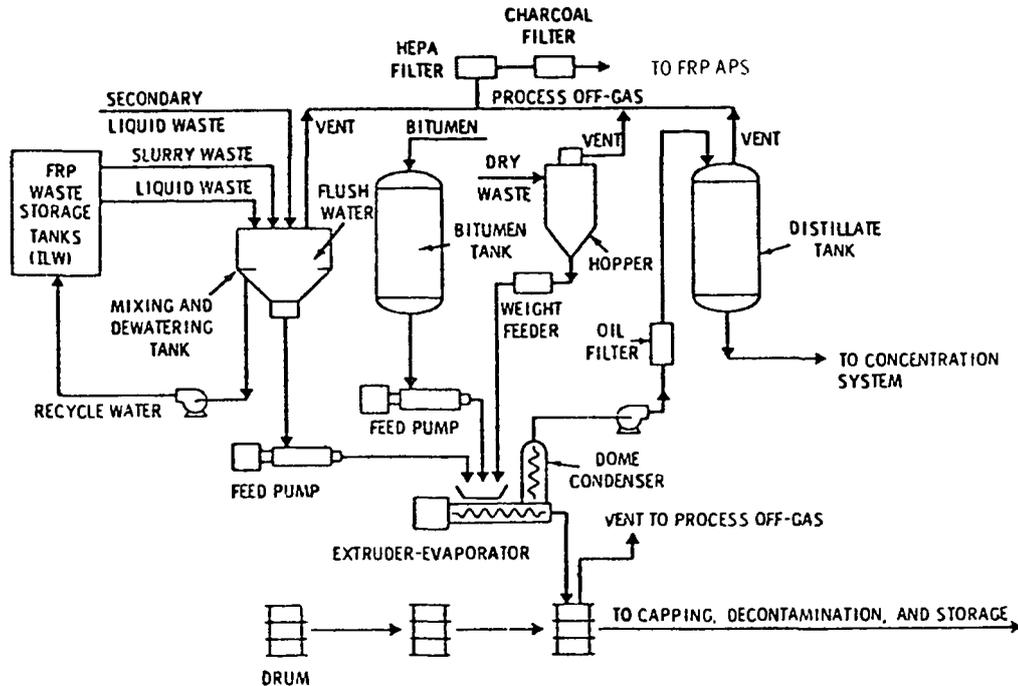


FIGURE 4.3.11. Process Flow Diagram for Bitumenization Facility at Fuel Reprocessing Plant

ruthenium may also be converted to a volatile species under normal process conditions. All of the other radionuclides present may also be present in off-gas and ventilation-air streams; these are present, however, as suspended particles rather than in a gaseous form. The fraction of the nonvolatile radionuclides suspended in the gas streams is generally quite small.

Gaseous and airborne wastes will have to be treated to remove radionuclides whether the spent fuel is discarded (the once-through case) or reprocessed. However, the complexity of treatment operations might vary widely depending on which cycle is chosen. The treatment operations will be at a minimum if spent fuel is packaged as intact assemblies (as in Section 4.3.1.1) and will be at a maximum if spent fuel is dissolved for disposal or reprocessing.

4.3.4.1 Filtration

Filtration is employed to remove radioactive particles from air streams being discharged from various equipment and facilities used in the LWR fuel cycle. Such particles arise from a variety of sources and mechanisms and their release to the environment can be controlled by a variety of filtration processes. There has been much experience in this area, since filtration has been successfully employed for many years in operating nuclear facilities.

One type of filter used almost universally in nuclear installations is the high-efficiency particulate air (HEPA) filter. These filters are composed of a specially formulated glass fiber web contained in a wood or metal frame. HEPA filters are available in several modular sizes; the size most commonly used for large installations is 61 cm on a

side by 29 cm deep. Strict quality assurance by the manufacturer and installer ensures that every filter will be at least 99.7% efficient for removing particles of 0.3 μm diameter. A 99.9% efficiency for removing radioactive particles (a decontamination factor (DF) of 10^3) is taken as a reasonable estimate for each stage of HEPA filtration. Higher removals are achieved by the use of multiple stages.

Prefilters are used to remove particles larger than 6 μm and have less efficiency for smaller particles. Prefilters are intended to remove the usual ambient dust from the air stream and thus double or triple the service life of the highly efficient HEPA filter. For radionuclide release calculations, a 91% efficiency for prefilters in removing radioactive particles (a DF of 10) is taken as a reasonable estimate.

Most nuclear facility designs include final filtration of essentially all of the air leaving the facility as well as prior filtration of the air leaving individual portions of the facility (e.g., some process equipment, cells, glove boxes). This is outlined in the flow diagram shown in Figure 4.3.12. The final filtration system has been termed the atmospheric protection system (APS). Three types of atmospheric protection systems are examined in detail in DOE/ET-0028 (Section 4.11) for application at fuel reprocessing plants (similar systems could be used at MOX-FFP and spent fuel treatment facilities). These three APS types use HEPA filters for final filtration but use different types of prefilters. One type of APS employs a commercially available Group III throw-away prefilter, another type employs a sand-bed prefilter, and the third type employs a deep-bed glass fiber filter. The Group III prefilter option was chosen as the example case in this Statement.

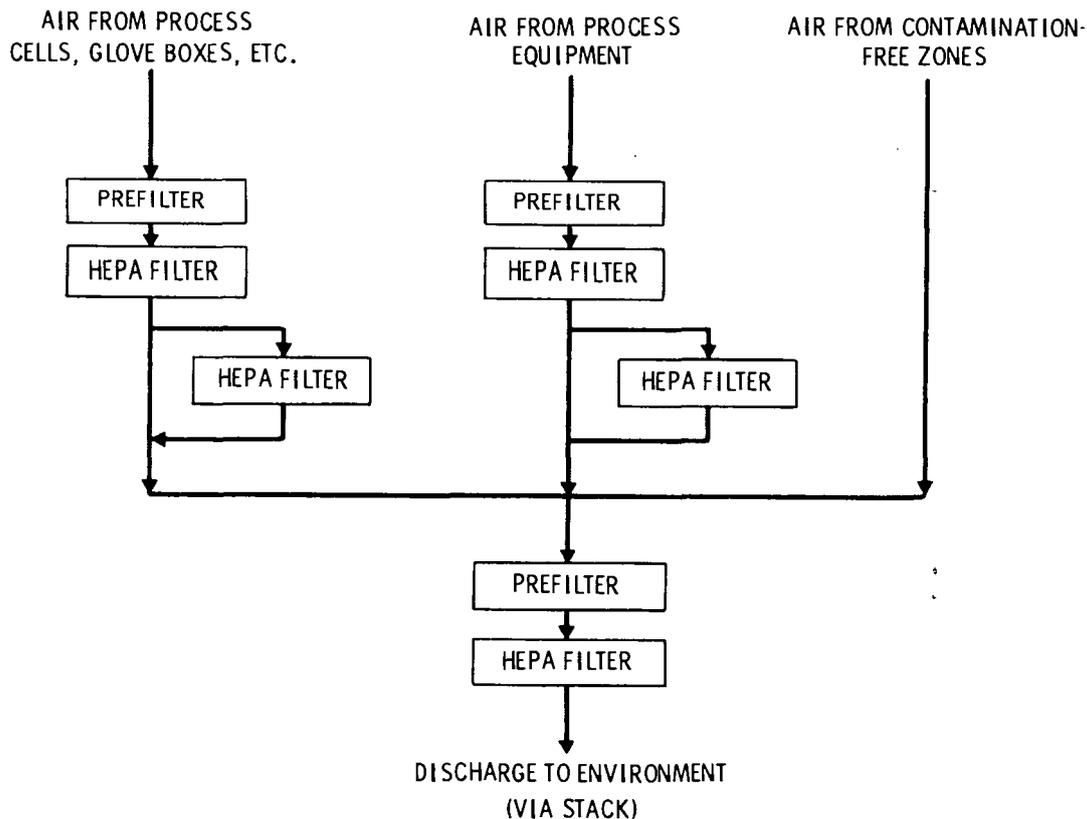


FIGURE 4.3.12. Flow Diagram for Filtration of Airborne Wastes

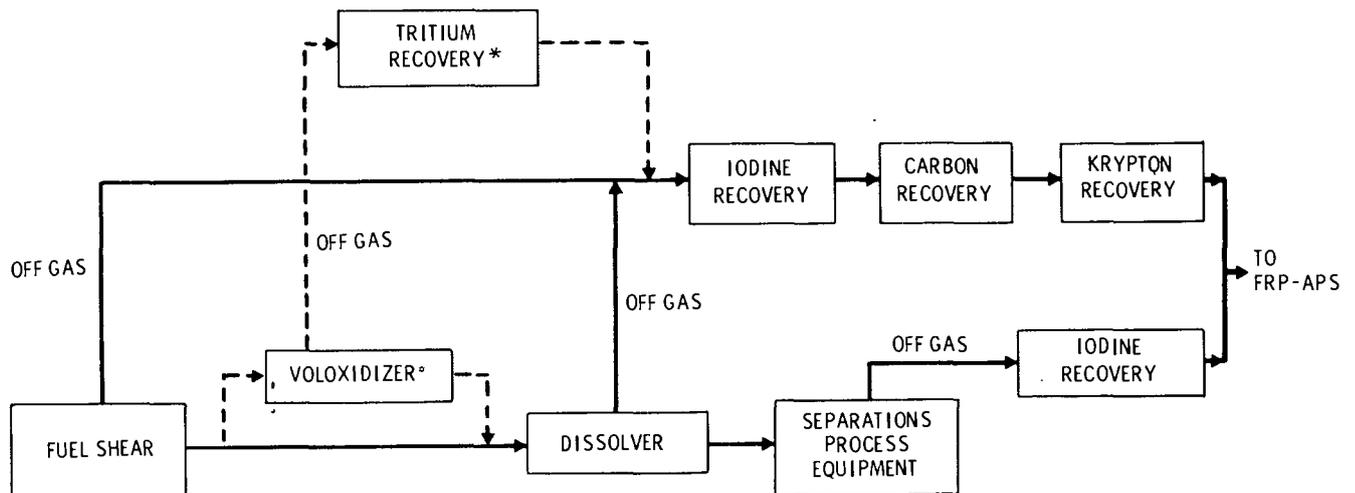
4.3.4.2 Gaseous Radionuclide Recovery

Where recovery of gaseous radionuclides (i.e., ^3H , ^{14}C , ^{85}Kr , ^{129}I) from airborne waste streams is required, processes other than filtration must be employed. Recovery of at least some of these gaseous radionuclides will be required if the spent fuel is processed to convert it to an alternative disposal form in the once-through case or to recover uranium and plutonium for recycle. In the example process of this Statement for the once-through case (the packaging of intact spent fuel assemblies), it is anticipated that no gaseous radionuclide recovery will be required. This is because only small quantities are expected to escape from the fuel.

Recovery of the gaseous radionuclides ^{14}C , ^{85}Kr , and ^{129}I (but not of ^3H) is included in the example off-gas treatment process used in this Statement for the reprocessing cycle. Most of this recovery takes place from the off-gas stream leaving the dissolver, since these radionuclides volatilize when the UO_2 fuel is dissolved in nitric acid. Iodine recovery from the gas streams leaving the separations process equipment is also provided, since a significant fraction of the iodine may remain in the dissolver solution and then volatilize later. Figure 4.3.13 presents a flow diagram for this gaseous radionuclide recovery system. The possible use of the voloxidation process to recover tritium is indicated also but, as mentioned previously, tritium recovery is not included in the example process of this Statement.

Tritium (^3H) recovery is not included in this Statement because the technology is not believed to have been suitably demonstrated as yet. In the example process, the tritium present in the UO_2 portion of the spent fuel is released to the atmosphere as water vapor. The bulk of this release occurs when the excess water is vaporized and discharged.

Methods of tritium control have been studied. The voloxidation process (Groenier 1977) has received the most development, but other alternatives have also been examined (Burger



* NOT INCLUDED IN THE EXAMPLE SYSTEM

FIGURE 4.3.13. Flow Diagram for Gaseous Radionuclide Recovery

and Scheele 1978). The voloxidation process involves oxidation of UO_2 to U_3O_8 at 400° to $500^\circ C$ in air. Essentially all of the tritium (plus portions of the other volatile radionuclides) is released to the gas stream by this process. The released tritium is removed from the gas stream (as water) by a bed of adsorbent material.

Although the example process in this Statement includes the recovery of three gaseous radionuclides, the study described in DOE/ET-0028 (Section 4.9) considered other possibilities as well. These included 1) no gaseous radionuclide recovery, 2) recovery of ^{129}I , 3) recovery of ^{129}I plus ^{14}C , and 4) recovery of ^{129}I plus ^{85}Kr .

In the example process, iodine recovery is effected by adsorption on silver zeolite, carbon recovery is accomplished by adsorption (as carbon dioxide) on zeolite molecular sieves, and krypton is recovered by cryogenic (very low temperature) distillation. Silver zeolite is prepared by replacing sodium ions in a zeolite with silver ions. Zeolite molecular sieves are crystalline aluminosilicates having pores of uniform size that completely exclude molecules which are larger than the pore diameter, thus permitting selective adsorption of those molecules that are smaller than the pore diameter.

The example off-gas treatment system also includes filtration for removal of particulate material, absorption and catalytic destruction steps for the removal of the oxides of nitrogen, NO and NO_2 , and ruthenium removal. A small portion of the ruthenium may be converted to a volatile form during processing operations. The example system uses beds of silica gel to remove this ruthenium before it reaches the processes used to recover the gaseous radionuclides.

The ruthenium-loaded silica gel and the iodine-loaded silver zeolite are ultimately disposed of in those forms; the estimated generation rates are 0.046 55-gallon drums/GWe-yr of the ruthenium waste (which requires remote handling) and 0.68 55-gallon drums/GWe-yr of the iodine waste. The carbon dioxide is desorbed from the molecular sieve and converted to solid calcium carbonate for disposal; 0.19 55-gallon drums/GWe-yr is the estimated quantity. The krypton-rich product (80% krypton and 20% xenon) from cryogenic distillation is collected in pressurized gas cylinders for storage; 2.8 cylinders/GWe-yr is the estimated quantity. These gas cylinders will require remote handling.

Alternatives exist for all of the processes employed in the example gaseous radionuclide recovery system. We do not mean to imply that the processes considered here are necessarily the best, only that they are representative of currently available technology. Krypton and carbon could be recovered by fluorocarbon absorption and iodine could be recovered by different solid sorbents or by scrubbing with various aqueous solutions. These alternatives have been discussed elsewhere (ERDA 1976).

4.3.5 Radionuclide Releases During Waste Treatment and Packaging

Estimates have been developed of radionuclide release during waste treatment and packaging operations in both the once-through and the reprocessing cycles. These estimates are summarized in Appendix 10A of DOE/ET-0028 for the packaging of intact spent fuel in a spent fuel packaging facility (SFPF) in the once-through cycle and for a variety of waste

treatment options at an FRP and at a MOX-FFP for the reprocessing cycle. Table 4.3.1 contains a summary of the releases estimated for radionuclides of potential importance during the treatment processes selected for use in this Statement. These release estimates are given as the fraction of the quantity present in spent fuel that is released during the treatment and packaging operations.

As mentioned earlier, tritium removal is not assumed in this Statement because the technology has not been fully demonstrated. Should the voloxidation process described earlier be successfully developed and applied, the release of tritium could be reduced to a value only 1% (or less) as large as that listed here.

All of these releases to the environment occur in gaseous or airborne waste streams. There are no planned discharges of radionuclide-contaminated liquid streams from these facilities.

4.3.6 Treated Waste Quantities

Table 4.3.2 contains a summary of the ranges of quantities of treated and packaged high-level, TRU, and gaseous wastes that result from implementation of various options of the once-through or reprocessing cycles described in Sections 4.3.1 through 4.3.4. These quantities are given in terms of the number of waste packages rather than in terms of the volume of waste because, for the mined geologic repository concepts used in this Statement, the repository area required for high-level waste is a function of the waste heat output while the area required for remotely handled TRU wastes is a function of the number of containers rather than of the volume of waste (see Section 5.3). The data for the packaging of intact fuel in the once-through case and for the packaging of the reprocessing wastes were taken from DOE/ET-0028. The data for the packaging of processed spent fuel were taken from ONWI-39.

TABLE 4.3.1. Estimated Radionuclide Releases During Waste Treatment and Packaging

Cycle	Waste Category	Facility	Release During Treatment and Packaging, Fraction of That in Spent Fuel ^(a)												
			Fission Products								Actinides			Activation Products	
			H	Kr	Sr	Ru	I	Cs	Ce	Pu	Am	Cm	C	Fe, Co, Ni	
Once-Through	Spent Fuel	SFPF	2×10^{-6}	6×10^{-5}	1×10^{-12}	1×10^{-12}	2×10^{-5}	4×10^{-11}	1×10^{-12}	0	0	0	6×10^{-6}	1×10^{-10}	
Reprocessing	High-Level Liquid Waste	FRP	8×10^{-2}	0	2×10^{-15}	1×10^{-10}	5×10^{-6}	2×10^{-15}	2×10^{-15}	1×10^{-17}	2×10^{-15}	2×10^{-15}	0	0	
	Fuel Residue	FRP	6×10^{-7}	0	2×10^{-16}	2×10^{-16}	0	2×10^{-16}	2×10^{-16}	2×10^{-16}	2×10^{-16}	2×10^{-16}	8×10^{-14}	5×10^{-13}	
	Failed Equipment and Noncombustible Waste	FRP	2×10^{-20}	0	2×10^{-15}	2×10^{-15}	6×10^{-23}	2×10^{-15}	2×10^{-15}	2×10^{-15}	2×10^{-15}	2×10^{-13}	2×10^{-20}	1×10^{-19}	
		MOX-FFP	0	0	0	0	0	0	0	2×10^{-14}	$5 \times 10^{-15(b)}$	0	0	0	
	Combustible Waste and Wet Wastes	FRP	2×10^{-6}	0	6×10^{-17}	3×10^{-15}	2×10^{-3}	6×10^{-17}	6×10^{-17}	3×10^{-14}	3×10^{-16}	3×10^{-16}	5×10^{-17}	3×10^{-16}	
		MOX-FFP	0	0	0	0	0	0	0	1×10^{-15}	2×10^{-14}	0	0	0	
	Gaseous and Airborne Primary Wastes	FRP	8×10^{-1}	1×10^{-1}	1×10^{-14}	2×10^{-8}	1×10^{-3}	1×10^{-14}	1×10^{-14}	2×10^{-11}	1×10^{-14}	1×10^{-14}	1×10^{-2}	0	
		MOX-FFP	0	0	0	0	0	0	0	1×10^{-12}	3×10^{-11}	0	0	0	
	Total Wastes from Reprocessing		9×10^{-1}	1×10^{-1}	1×10^{-14}	2×10^{-8}	3×10^{-3}	1×10^{-14}	1×10^{-14}	1×10^{-14}	3×10^{-11}	1×10^{-14}	1×10^{-2}	5×10^{-13}	

(a) Quantities present in spent fuel are listed in Tables 4.4.2 and 4.2.4.

(b) Assuming reprocessing 1.5 years after reactor discharge and fuel fabrication one year later.

TABLE 4.3.2. Estimated Quantities of Packaged High-Level, TRU, and Gaseous Wastes

Packaged Waste	Package Type	Packages/GWe-yr					
		Intact Fuel ^(a)	Once-Through Case		Reprocessing Case		
			Processed Fuel ^(b)		Example	Low	High
			Low	High			
High-Level							
Spent fuel	Canister	127	61	141	---	---	---
Solidified Liquid Waste	Canister	---	---	---	35	27 ^(c)	44 ^(c)
Remotely Handled							
Fuel Residue	Canister	---	12	29	9.1	3.7	9.1
Failed Equipment	Canister	---	2	3	1.4	---	---
	Drum	---	---	---	9.0	---	---
Compressed Gas	Canister	---	0.3	0.4	---	---	---
	Gas cylinder	---	---	---	2.8	0	2.8
Other	Canister	---	28	43	---	---	---
	Drum	---	---	---	146	130	316
Contact Handled							
Failed Equipment	Box	---	---	---	1.5	---	---
Other	Drum	---	6.5	9.4	93	29	281
Total		127	110	226	298	190	653

(a) The example case described in Section 4.3.1.1.

(b) For the cases described in Sections 4.3.1.2 and 4.3.1.3.

(c) For canister heat loadings of 1.2 to 3.2 kW, assuming 6.5 years after reactor discharge.

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4.4 WASTE STORAGE

The treated and packaged wastes (Section 4.3) may have to be stored for an interim period of time before they are finally placed in a repository. With some wastes (e.g., spent fuel in the once-through case and high-level waste in the reprocessing cycle case), interim storage is desirable to allow many of the radionuclides to decay; this lowers the rate of heat generation and simplifies the final disposal operations. With other wastes, there is no technical reason for storage prior to disposal, but storage may be required while awaiting availability of a final repository. With yet another type of waste (krypton), a special facility may be required to store the waste until its radioactivity has decayed to a level low enough that it can be released.

4.4.1 Spent Fuel Storage

Storage of spent fuel is an integral part of both the once-through and the reprocessing cycles. In both cases, an initial storage period is aimed at allowing short-lived radionuclides to decay away; this results in a lowered heat generation rate that facilitates subsequent handling operations and also reduces the degree of radionuclide containment required during the processing operations. Unpackaged spent fuel has been stored in water basins in the U.S. for many years. The initial storage period was first envisioned as lasting only about one year, after which the fuel would be reprocessed. However, because of deferral of reprocessing and the possibility that spent fuel may be sent to disposal without reprocessing, and thus require storage until a repository is available, the initial storage period may now last 20 years or more.

Even longer storage before disposal or reprocessing may be desirable or necessary. Thus, extended (up to 100 years) storage of spent fuel has also been examined. Advantages include additional reductions in the radionuclide heat generation rate and the continued availability of the fuel if the decision is made to reprocess spent fuel.

The extended storage concepts examined here involve prior packaging of the fuel, as described in Section 4.3.1.1, although it could well be that water basin storage of unpackaged fuel would be satisfactory for this purpose also. Only intact spent fuel is considered here for extended storage; it is assumed that if spent fuel is to be processed to a different form for disposal, the processing would not be done until the time of disposal. Four storage modes for packaged intact spent fuel are described briefly here along with the water basin storage of unpackaged spent fuel. More detailed descriptions are presented in DOE/ET-0028, Section 5.

Water basin storage is the only method considered in this Statement for unpackaged spent fuel. The four packaged fuel storage concepts are described here to illustrate the range of alternatives available to reduce the already negligible impacts of spent fuel storage to even lower values.

4.4.1.1 Water Basin Storage of Unpackaged Spent Fuel (Example Method)

The storage of spent power reactor fuel in water basins is an established technology that has been used successfully for over 20 years. Water basin storage has been employed at government-owned reactors and commercial light water reactors, fuel storage basins, and a fuel reprocessing plant. The water basin storage of unpackaged spent fuel at independent spent fuel storage facilities and at stand-alone at-reactor basin facilities is discussed in more detail in separate environmental impact statements (DOE/EIS-0015 1980 and NUREG-0575 1979). Water basin storage at independent spent fuel storage facilities was also examined in detail in DOE/ET-0028.

Spent fuel elements arrive at independent storage facilities in shipping casks. The elements are removed from the casks and are placed in storage baskets (containers) that are designed to separate the fuel assemblies sufficiently to assure criticality safety. The baskets are then moved to pool storage positions.

During water basin storage, the pool water serves both as a radiation shield and a heat transfer medium to remove the radionuclide decay heat. This heat is then dissipated to the atmosphere via a cooling tower by means of a secondary (and separate) recirculating cooling system. The water quality in the pool is maintained by filtration and ion exchange.

Two independent water basin storage facilities for unpackaged spent fuel are described in DOE/ET-0028 (Section 5.7). One facility stores LWR fuel assemblies containing 3000 MTHM (metric tons of heavy metal) in six pools (each with a storage capacity of 500 MTHM) and has the capability to receive and/or ship spent fuel at a rate of 1000 MTHM/yr. The other facility is similar but is modified to receive spent fuel at a higher rate and route it to an adjacent fuel packaging facility. This modified facility has the capacity to receive spent fuel at a rate of 2000 MTHM/yr and to store spent fuel containing 3050 MTHM. Other sizes are considered in DOE/EIS-0015.

Radionuclide emissions during operation of such facilities were estimated for receiving and shipping operations and for the storage condition. Table 4.4.1 contains these estimates. These radionuclide emissions occur via the gaseous and airborne release route; no aqueous releases containing radionuclides are expected.

4.4.1.2 Water Basin Storage of Packaged Spent Fuel

The water basin storage of packaged spent fuel is similar to that for unpackaged fuel except that the fuel elements are placed into stainless steel canisters before storage. Packaging of intact spent fuel was discussed in Section 4.3.1.1. These canisters provide additional fuel protection, radionuclide containment barriers, and contamination control.

The facility for water basin storage of packaged spent fuel (see DOE/ET-0028, Section 5.7.5) is somewhat different from that for storage of unpackaged fuel. Each packaged fuel pool is designed to store spent fuel containing 2000 MTHM. The facility is designed for modular expansion to a total of ten such pools for a storage capacity of 20,000 MT.

TABLE 4.4.1. Estimated Radionuclide Releases During Water Basin Storage of Unpackaged Spent Fuel

<u>Fission Products</u>	<u>Fraction(a) Released During Receiving or Shipping</u>	<u>Fraction(a) Released Each Year During Storage</u>
H	2×10^{-6}	1×10^{-6}
Kr	6×10^{-5}	7×10^{-7}
I	1×10^{-7}	9×10^{-9}
Cs	7×10^{-11}	9×10^{-12}
All Others	2×10^{-12}	2×10^{-13}
<u>Actinides</u>	Negligible	Negligible
<u>Activation Products</u>		
C	3×10^{-6}	1×10^{-8}
All Others	2×10^{-10}	2×10^{-11}

(a) Fraction of activity in spent fuel released to atmosphere. See Tables 4.2.2 and 4.2.4 for the activity in spent fuel.

The radionuclide emissions from a facility storing packaged fuel will be markedly lower than those from a facility storing unpackaged fuel. The radionuclide emissions are assumed to be negligible since the containment of the fuel elements in high-integrity packages will reduce the emissions by at least several orders of magnitude below the already low releases resulting from storage of unpackaged fuel.

4.4.1.3 Air-Cooled Vault Storage of Packaged Spent Fuel

Another alternative for extended storage of packaged fuel involves packaging in carbon steel canisters and storing in heavily shielded, air-cooled concrete vaults. The conceptual facility (see DOE/ET-0028, Section 5.7.6), is an adaptation of a storage concept for solidified high-level waste (ARHCO 1976). In this concept natural-draft air circulation is used to remove decay heat so that no mechanical equipment is required for heat removal. The spent fuel canisters are placed vertically within steel sleeves in the vault; these sleeves increase the natural air flow velocity around the canisters and provide additional heat transfer area for the air coolant. Air enters a bottom plenum through side inlets in the structure, passes upward through annuli formed by the storage units and sleeves, and is discharged through an exhaust port to the atmosphere. Air flow is induced by the decay heat of the spent fuel and the design of the vault. This concept has not been used for fuel storage, but is based on established engineering practice and principles.

Double containment of the radionuclides maintains radionuclide emissions at negligible levels. Double containment is provided by single encapsulation of unfailed fuel assemblies (cladding is one barrier and the canister is the second) and by double encapsulation of failed fuel assemblies. A more conservative approach would be to doubly encapsulate all of the assemblies.

The exhaust air is monitored to provide early detection of emissions. If container failure is indicated, the contaminated air is diverted through an adjacent sand filter by forced draft exhaust blowers. The failed package is removed to a facility for repackaging or overpacking. Package failure is expected to be rare or non-existent.

Each sleeve contains either four PWR or nine BWR individually packaged fuel assemblies. The referenced design provides for 1120 sleeves per storage vault and for modular expansion up to a total of ten vaults. Each vault would store spent fuel containing 2000 MTHM, for a total storage capacity of 20,000 MTHM.

4.4.1.4 Dry Well Storage of Packaged Spent Fuel

The concept of dry wells (also called dry caissons) for the storage of packaged spent LWR fuel is similar to concepts already in use for other reactor fuels in both the U.S. (Hammond et al. 1971) and in Canada (Morrisen 1974). For the conceptual facility here (see DOE/ET-0028, Section 5.7.7), the spent fuel is packaged in carbon steel canisters and placed in an underground steel- and concrete-lined caisson. The caisson is then closed with a concrete plug. This concept relies upon the soil to conduct the decay heat from spent fuel to the earth's surface, where it is dissipated to the atmosphere. As in the other packaged fuel storage concepts, double containment is depended on to maintain radionuclide releases at negligible levels.

The caisson is designed so that its atmosphere may be monitored and sampled periodically. Water run-off from the storage area will be collected and monitored (and decontaminated, if necessary) before release. Package failure is considered a highly unlikely event; should it occur, the package is returned to the packaging facility for repackaging or overpacking.

Each caisson provides a storage space of about 1 m in diameter by 5 m high and contains either three PWR or six BWR individually packaged fuel assemblies. The design provides for incremental expansion up to 15,800 caissons, which would store spent fuel containing 20,000 MTHM.

4.4.1.5 Surface Cask Storage of Packaged Spent Fuel

In the surface cask storage concept, packaged spent fuel is stored (outdoors) in a reinforced concrete radiation shield (cask). This concept has been extensively studied (ARHCO 1976) and is a straightforward application of existing technology. In the variation described (see DOE/ET-0028, Section 5.7.8), spent fuel assemblies in carbon steel canisters are placed in vertical concrete casks located outdoors on concrete pads. Heat is removed from the fuel by natural convection air flow upward through the annulus between the cask and the fuel packages.

As in the other packaged fuel storage concepts, double containment limits radionuclide emissions to negligible levels. Monitoring capability is provided to detect radionuclide leakage and also to detect increases in exit air temperature, which would indicate blockage

of air ports. Failed packages would be returned to the packaging facility for canister repair or replacement, as necessary; this is considered to be an improbable event.

Each storage unit is about 3.3 m (10 ft) in diameter and about 7.6 m (25 ft) high. Each unit provides a storage envelope of about 1 m in diameter by 5 m high, and contains either four PWR or nine BWR individually packaged fuel assemblies. A large number of storage units would be located at one site; the referenced design provides for incremental expansion up to a total of 11,200 storage units, which would store spent fuel containing 20,000 MTHM.

4.4.2 High-Level Waste Storage

In the reprocessing cycle case where the fuel to be reprocessed has been out of the reactor only a few years, the storage of high-level waste either as a liquid or a solid is desirable to provide additional time for the heat generation rate to decrease. Another potential reason for storage of high-level waste could be to bridge the (possible) gap between waste generation and repository availability. The high-level waste could be stored as a liquid and then be solidified just before repository emplacement, or it could be solidified immediately and then stored in that form until it could be placed in a repository, or it could be stored as a liquid for part of the time and as a solid for part of the time (although the latter case would doubtless be more expensive).

Except for moderate volumes of surge storage in shielded processing facilities, the only method given serious consideration anywhere for interim storage of liquid high-level waste is storage in large underground tanks. Many methods, however, appear suitable for storage of high-level waste after it has been solidified. Solidified high-level waste packages can be stored similarly to spent fuel in water basins, in air cooled vaults, in dry wells, and in casks stored on the surface (ERDA-76-43 1976). Additional details on the storage of liquid high-level waste and of solidified high-level waste in water basins and in sealed casks can be found in DOE/ET-0028 (Section 5).

In the example waste management system considered in this Statement for the reprocessing cycle case, spent fuel is reprocessed 1.5 years after discharge from the reactor. The resultant high-level liquid waste is solidified immediately (except for a minimal surge storage period) and the solidified high-level waste is stored for 5 years in a water basin at the reprocessing plant. When further storage is required pending repository availability, the waste is stored in sealed casks. Certain other waste disposal concepts under consideration (i.e., rock melting and well injection) dispose of high level waste as a liquid. Implementation of one of these concepts may require substantial liquid high-level waste storage facilities.

4.4.2.1 Tank Storage of Liquid High-Level Waste

Storage of liquid high-level waste in large subsurface tanks has been practiced for over 30 years in several countries. Most of the U.S. experience has involved storage of government-produced defense program wastes; the tanks built initially were single-walled,

but double-walled tanks have been built in recent years at both Hanford and Savannah River to reduce the possibility of leakage of waste into the environment (DOE/EIS-0063 1980 and DOE/EIS-0062 1980). The defense program wastes were neutralized before storage (by the addition of hydroxides) and are stored in carbon steel tanks. The commercial wastes produced at the West Valley Plant in New York are also stored in this way. More recent plans involve storage of acidic waste in stainless steel tanks. Such tanks have been built (but not used) at the Barnwell Plant in South Carolina. The design concept here (see DOE/ET-0028, Section 5.1) is similar to that used at Barnwell.

The tanks employ double containment, consisting of a primary stainless steel container within a stainless steel liner. Both containers are supported by and encased in a reinforced concrete vault. The tanks in this design are 17 m (54 ft) in diameter and 6 m (20 ft) high and have a net storage volume of 1140 m³ (300,000 gal) with 10% freeboard. Each such tank has the capacity to store the concentrated high-level liquid waste resulting from reprocessing spent fuel containing 2000 MTHM. Seven tanks are required to provide capacity for 5-yr storage of the high-level waste produced at a 2,000 MT fuel reprocessing plant (four tanks filled, one filling, one emptying and one tank held as a spare). The radioactive decay heat is removed by cooling water, which passes through coils installed in the tanks; the heat is then dissipated via a cooling tower. The contents of the tank are continuously mixed by airlift circulators and by ballast tanks that provide an intermittent flushing action.

The tank off gases are treated to remove any volatilized iodine and particulate radionuclides that might be entrained in the gas stream. Estimated radionuclide emissions are given in Table 4.4.2.

TABLE 4.4.2. Estimated Radionuclide Releases During Tank Storage of Liquid High-Level Waste

<u>Fission Products</u>	<u>Fraction(a) Released Each Year During Storage</u>
H	8×10^{-3}
Kr	0
I	5×10^{-7}
Ru	1×10^{-12}
All Others	1×10^{-13}
<u>Actinides</u>	
U	5×10^{-16}
Pu	5×10^{-16}
All Others	1×10^{-13}

(a) Fraction of activity in spent fuel released to atmosphere. See Table 4.2.4 for the activity in spent fuel.

4.4.2.2 Water Basin Storage of Solidified High-Level Waste (Example Method)

Solidified high-level waste packages (described in Section 4.3.2) can be stored in water basins in much the same manner as that described in Section 4.4.1.1 for the water basin storage of spent fuel. In the facility for water basin storage of solidified high-level-waste examined here (see DOE/ET-0028, Section 5.4.1), the singly encapsulated (in stainless steel) waste is received for storage from an adjacent waste solidification facility. The waste canisters are stacked in double-tiered racks in water basins, each of which is designed to hold the waste from reprocessing spent fuel containing 1,500 MTHM. Each basin is equipped with a water purification system and a heat exchanger system to remove the decay heat, which is dissipated to the atmosphere via a cooling tower. Eight such basins are included in the facility design. Radionuclide emissions estimated for water basin storage of vitrified high-level waste are given in Table 4.4.3.

4.4.2.3 Sealed Cask Storage of Solidified High-Level Waste

The sealed storage cask concept for extended storage of solidified high-level waste involves encapsulating the waste canister in a high-integrity, sealed metal storage cask and then placing the doubly encapsulated waste in a reinforced concrete radiation shield. The assembly is then placed on a base in a large outdoor storage yard. Air circulates by natural convection between the radiation shield and the sealed cask to remove the heat being generated by the waste. This concept has been studied extensively (ARHCO 1976).

A facility to implement this concept was designed to accommodate 0.3 x 3 m waste canisters generating about 4.4 kW of decay heat (see DOE/ET-0028, Section 5.4.2). The facility's initial capacity is 2,000 canisters of waste; it can be expanded in 2,000 canister modules to an ultimate capacity of 20,000 canisters.

TABLE 4.4.3. Estimated Radionuclide Releases During Water Basin Storage of Vitrified High-Level Waste

<u>Fission Products</u>	<u>Fraction^(a) Released Each Year During Storage</u>
H	0
Kr	0
I	0
Cs	2×10^{-13}
All Others	2×10^{-14}
<u>Actinides</u>	
U	1×10^{-16}
Pu	1×10^{-16}
All Others	2×10^{-14}

(a) Fraction of activity in spent fuel released to atmosphere. See Table 4.2.4 for the activity in spent fuel.

The storage yard is monitored to detect any radionuclide leakage from the storage units. Radionuclide emissions are assumed to be negligible since leakage of the doubly encapsulated waste is believed to be highly improbable. Canisters that do leak can be retrieved and repackaged.

4.4.2.4 Other Solidified High-Level Waste Storage Concepts

Solidified high-level waste could be stored in an air-cooled vault facility similar to that described in Section 4.4.1.3 for the storage of spent fuel. In fact, the conceptual facility for spent fuel storage is an adaptation of a concept for storage of solidified high-level waste (ARHCO 1976). Double containment of the radionuclides in the high-level waste could be provided by overpacking the primary canister. The design for a solidified waste facility would be tailored to the high-level waste canister size and heat generation rate.

Dry well storage of solidified high-level waste could also be employed. This would resemble the dry well storage of spent fuel described in Section 4.4.1.4. Well size and spacing would be different for the solidified waste than for the spent fuel, depending on waste canister size and heat generation rate. Double containment of the waste by overpacking the primary canister could also be utilized for this storage concept.

4.4.3 TRU Waste Storage

The packages of treated TRU waste described in Section 4.3.1 for the once-through case and in Section 4.3.3 for the reprocessing case could require storage for an interim period before a repository is available.

The packaged wastes are considered in one of two categories depending on the radiation level. Packages that have surface dose rates no higher than 200 millirem/hr are "contact-handled," i.e., workers can handle them without extensive shielding. Packages with higher surface dose rates require shielding and/or remote handling to protect operating personnel; these packages are "remotely handled."

The TRU waste packages with the highest surface dose rates are the canisters containing the fuel residues (the fuel hulls and hardware). Some disassembled failed equipment is also assumed to be packaged in identical canisters. Two alternative interim-storage facility concepts for these canisters are described here (see also DOE/ET-0028, Section 5.2): vault storage and dry-well (near-surface) storage. The dry well concept is used as the example method in this Statement.

Other remotely handled TRU wastes are packaged in steel 55-gal drums. Vault storage and dry well storage facility concepts for these wastes are described here (see also DOE/ET-0028, Section 5.3). Vault storage is used as the example method in this Statement.

The contact-handled wastes are packaged in steel boxes or drums. Unshielded indoor storage and outdoor surface storage facility concepts are described for these wastes. The outdoor surface storage concept is the example concept used in this Statement.

Because of the lower radionuclide content and the integrity of the waste packages, no significant releases of radionuclides are anticipated from any of these conceptual TRU waste storage facilities. However, effluents would be monitored to verify that this is indeed true and to provide early detection of problems that might arise.

4.4.3.1 Vault Storage of RH-TRU (Example Method for Drummed RH-TRU)

In the vault storage concept for remotely handled wastes, the waste is considered to be packaged either in special canisters (0.76 m dia x 3 m) or in 55-gal drums. Vault storage is the example concept of this Statement for these 55-gal-drum-packaged wastes and an alternative concept for these canistered wastes.

The 55-gal drums that require remote handling are simply stacked in cells constructed of reinforced concrete. The drums are unloaded from the shipping container and are placed in the storage cells by a crane using a vacuum-operated lifting device. The design calls for each cell to contain 500 drums; these are five layers of drums, 100 drums in each layer, and plywood sheets separate the layers. The basic storage module contains 40 such cells holding a total of 20,000 drums. Facility designs were evaluated for storage both at an individual fuel reprocessing plant and at an independent site serving a number of reprocessing plants.

The vault storage concept for the canistered waste uses individual sleeves for canister storage in concrete vaults, which provide radiation shielding. The canisters are handled with a remotely operated crane. They are lowered from shipping casks through a special transfer device into the storage space and a shielding plug is placed above the canister. Each storage space is a galvanized steel pipe (0.9 m in dia) with a plate welded to the bottom and is suspended from the roof slab of the vault. Natural air circulation through the vault provides canister cooling. The vault storage concept for canisters is based on a modular design. Each cell has a capacity of 312 canisters. Facility designs were evaluated for siting both at an individual fuel reprocessing plant and at an independent site serving a number of reprocessing plants.

4.4.3.2 Dry-Well Storage of RH-TRU (Example Method for Canistered RH-TRU)

The dry-well storage concept, which is the example concept of this Statement for the storage of canisters containing the fuel residue and some of the failed equipment, involves construction of storage spaces in an above-grade soil structure (berm). The canisters are placed in individual storage spaces positioned vertically in the berm, and the spaces are capped with steel and concrete plugs. The plug, canister, and shipping cask are handled remotely using a crane. Each storage space consists of a galvanized steel pipe sleeve (0.9 m in dia) with a plate welded to its bottom and suspended from a slab; gravel is back-filled around the outside of the pipe. Heat is removed by conduction through the soil to the atmosphere. The basic module designed for the dry-well storage of canisters has two berms, each containing 1,248 storage spaces.

A similar approach was examined as an alternative for the storage of the waste packaged in 55-gal drums that requires remote handling. In this instance 5 drums are stored in each caisson (0.66 m dia x 5.2 m deep). Most of the drums can be unloaded from the shipping container and placed in storage using only a shielded mobile yard crane that has a vacuum lifting device. Drums having high surface dose rates are transferred to the caisson using a bottom loading cask. In this design, 504 storage spaces are provided in each module.

4.4.3.3 Unshielded Indoor Storage of CH-TRU

The packages of TRU waste that can be contact-handled can be stored indoors in an unshielded facility. A conceptual facility examined as an alternative to outdoor storage consists of a precast concrete building containing a number of individual storage cells. Drums (55-gal) are stacked six high in horizontal layers; plywood sheets are placed between the layers. Steel boxes are also used to package such wastes; a storage box occupies the space of 12 drums. The boxes and drums are handled by mobile cranes and by fork-lift trucks.

The basic module used in this design includes two cells, each of which will store 4,200 drums. When storage capacity beyond that provided by the basic module is required, an expanded version of the basic module is used or multiples of the basic module are employed.

4.4.3.4 Outdoor Storage of CH-TRU (Example Method)

Outdoor storage is the example concept of this Statement for contact-handled TRU wastes. This approach is presently used at most government installations. Several variations are in use, involving below-grade as well as above-grade techniques and differing amounts of weather protection. The most widely accepted method is to place the waste packages on some structural pad, and cover them first with an impermeable membrane, and then with dirt.

In this design the drums and boxes of waste are placed on an above-ground asphalt slab that is contained within a temporary air-supported structure to allow operations to continue during inclement weather. The containers are arranged in horizontal layers; sheets of plywood are placed over each layer before the next layer is added. Handling of the containers is by mobile crane and by a drum grabber. As the storage area is filled, polyethylene sheets are placed over the stacked containers and the stack is covered with dirt to a depth of at least 0.9 m. Once a storage area is completely filled and covered with earth, the air-supported structure is removed, and the dirt cover is either seeded or covered with a bitumen layer.

The basic storage module for this concept has a storage capacity for 10,000 55-gal drums of waste. Capacity can be expanded by either using an expanded version of the basic module or by using multiples of the module.

4.4.4 Krypton Storage

The ^{85}Kr removed from the off-gas stream as described in Section 4.3.4.2 must also be stored. This gaseous radionuclide can be encapsulated and stored in pressurized gas cylinders. Alternative krypton encapsulation techniques being investigated include 1) zeolite encapsulation, where krypton is diffused into "crystalline cages" at high temperatures and pressures, and where escape of the krypton is slow at low temperatures; 2) dissolution in a glass matrix, where krypton is trapped within a glass when it solidifies; and 3) entrapment of krypton in metal solids during high-rate sputtering.

The krypton storage facility chosen for this Statement stores gas cylinders containing about 80% krypton and 20% xenon. The radionuclide heat generation rate from such cylinders is appreciable and refrigerated air cooling is provided. The surface dose rates of the cylinders are such that remote handling is required; this is provided by special transfer containers and cranes.

The storage plan for krypton differs from those for the other wastes in an important respect. Since the half-life of ^{85}Kr is relatively short (10.7 yr), it is assumed that after storage for 50 years or so the ^{85}Kr can be released. In 50 years the amount of ^{85}Kr remaining will be only 4% of the initial amount; after 60 years only 2% will remain.

The krypton storage facility (see DOE/ET-0028, Section 5.6) is located adjacent to a fuel reprocessing plant and is sized to handle the output of the plant during its lifetime. Separate storage cells, each holding 104 cylinders, are provided. The number of cells is increased every ten years to provide the necessary storage capacity; 14 cells are required for each ten years' output. The facility also includes hot cells for use in cylinder inspection and gas transfer (e.g., from a leaking cylinder to a sound cylinder) operations.

The gas cylinders are passed into the storage cell through ball valves and rest horizontally on shelves within the cell. Each storage cell contains five shelves and is provided with a self-contained air circulation and heat removal system. These air circulation systems are monitored to provide detection of leaks. If a minor leak is detected, the cylinder is sent to the hot cell and the contents are transferred to a new cylinder. If a cylinder suddenly ruptures, the cell atmosphere will be pumped to a holding tank where it will be sampled and then either returned to the fuel reprocessing plant or sent to the storage facility stack for release.

The normal release of ^{85}Kr from the storage facility occurs in two ways: 1) the small leakages from a number of cylinders, and 2) the planned discharge of the krypton at the completion of the storage period. The former release is estimated to amount each year to no more than 0.1% of the amount of ^{85}Kr present during the year. The latter release does not begin until completion of the planned storage period. For a 50-yr storage period, this release amounts to 4% of the amount initially placed into storage. The planned storage period (and, thereby, the planned release) can be changed after storage has begun.

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4.5 WASTE TRANSPORT

For the example once-through cycle, the waste transportation of concern for this Statement is the shipment of spent fuel. Other wastes would be non-TRU wastes that are not covered in this Statement. The spent fuel may be shipped directly from the nuclear power plants to an encapsulation facility located at the geologic repository site, or it may be shipped first to an interim storage facility and then to the encapsulation facility.

For the reprocessing cycle, transportation is considered for spent fuel, solidified high-level waste, and TRU wastes. Spent fuel may be shipped from the reactors either to interim storage or directly to reprocessing. Reprocessing plant and MOX fabrication plant waste packages may be shipped directly from the fuel reprocessing plants and from the mixed oxide fuel fabrication plants to the geologic repository, or they may be shipped first to an interim storage facility and then to the geologic repository.

The transportation of these wastes is discussed briefly in the following sections. More detail is contained in Section 6 of DOE/ET-0028.

4.5.1 Spent Fuel Transport

Spent fuel has been shipped in the United States for many years. Massive, heavily shielded shipping casks are available for both truck and rail transport of spent fuel from current-generation LWRs. Most spent fuel casks will accept either PWR or BWR spent fuel by using different fuel baskets; however, some are designed only for a particular fuel type. Table 4.5.1 gives information about casks that are currently available or licensed for spent fuel shipments in the U.S. More detailed information is contained in Sections 6.2.1 and 6.2.2 of DOE/ET-0028 and in Volume 2, Appendix C of DOE/EIS-0015.

TABLE 4.5.1 Available Shipping Casks for Current Generation LWR Spent Fuel

Cask Designation	Number of Assemblies		Approximate Loaded Cask Weight, MT	Usual Transport Mode	Shielding		Cavity Coolant	Maximum Heat Removal, kW	Number Available (a)
	PWR	BWR			Gamma	Neutron			
NFS-4 (NAC-1)	1	2	23	Truck	Lead and steel	Borated water and antifreeze	Water	12	7
NLI 1/2	1	2	22	Truck	Lead, uranium and steel	Water	Helium	11	5
TN-8	3		36	Truck (b)	Lead and steel	Borated solid resin	Air	36	2
TN-9		7	36	Truck (b)	Lead and steel	Borated solid resin	Air	25	1
IF-300	7	18	63	Rail (c)	Uranium and steel	Water and antifreeze	Water	76 (d)	4
NLI 10/24	10	24	88	Rail	Lead and steel	Water	Helium	97 (e)	2

(a) According to Winsor, Faletti, and De Steese (1980).

(b) Overweight permit required.

(c) Truck shipment for short distances with overweight permit.

(d) Licensed decay heat load is 62 kW.

(e) Licensed decay heat load is 70 kW.

These existing casks were designed to transport short-cooled (6 months or less) irradiated fuel, consistent with the earlier expectation of rapid recycling of fissile materials. The current situation, however, indicates that most spent fuel transport will involve fuel that has been cooled for at least several years. Consequently, there appears to be considerable incentive to build a fleet of casks specifically designed for this long-cooled fuel because its lower thermal and radiation output would permit an increase in cask capacity and a reduction in handling costs. Several cask fabricators have announced new cask construction programs; some of these address the prospect of transporting long-cooled fuel.

Existing cask designs are for the transportation of unpackaged spent fuel. Transportation of spent fuel that has been packaged in canisters (either as intact spent fuel or as treated spent fuel) will require some additional design modifications. If existing casks or cask designs cannot be suitably modified, new cask designs may be required.

Past experience indicates that an estimated six to eight years could be required to design, test, license, and then fabricate a fleet of newly designed casks. However, with a licensed standard cask, a vendor could significantly shorten the length of time required to deliver a fleet of casks. The useful life of spent fuel shipping casks is estimated to be 20 to 30 years.

Several factors can influence the choice of rail or truck casks for use in the shipment of spent fuel. Rail casks have a significantly larger payload than truck casks. About 10 times as much fuel can be shipped in a rail cask with an increase in shielding weight of only about a factor of 4 over the amount required for a truck cask. On the other hand, truck shipments normally require less time for completion than rail shipments. About 50% of the reactors now operating in the U.S. or scheduled for completion by 1980 do not have rail spurs at the site. Many of these reactors without rail spurs can be serviced by intermodal (truck or rail) casks, which require overweight permits for shipment by truck to the nearest rail siding.

In this Statement, it is assumed that 90% of unpackaged spent fuel will be shipped from reactors by rail and 10% by truck. To accommodate the reactors without rail access, half of the rail shipments are assumed to be in intermodal casks that allow truck shipment for short distances. Shipments from interim storage to repositories or reprocessing are assumed to be 100% by rail. Any shipments of packaged spent fuel are assumed to be by rail using casks that can handle 7 PWR or 17 BWR packaged assemblies. Spent fuel in the once-through cycle is assumed to cool at least five years before shipment. In the assumed reprocessing cycle, however, spent fuel (which is not a waste in this cycle) can be shipped to a reprocessing plant after one year cooling.

Transport of spent fuel by barge and by ship has also been considered. Barge transport is an alternative when both the nuclear power plant and the encapsulation or storage facility are on navigable waterways. Barge transport suggests high payloads and low tariffs. However, cost gains in these two areas could be offset by the longer transit times estimated for barge shipments. Should offshore (floating) nuclear power plants be constructed, barge transport is an obvious choice for the initial portion of the journey of the

spent fuel to an encapsulation or storage facility. Casks for barge shipment of spent fuel would probably be similar, if not identical, to those used for rail transport.

Ship transport of spent fuel could be required if some of the alternatives to geologic disposal (e.g., island, subseabed, icesheet) described in Chapter 6 of this Statement are implemented. Casks for spent fuel transport by ship would probably require adaptation or modification of existing design. The design would likely vary somewhat depending on the specific disposal concept, but could be similar to those of existing casks.

4.5.2 High-Level Waste Transport

High-level waste transport is required in the example reprocessing cycle. Solidified high-level waste could be shipped in specially designed casks by truck, rail, barge, or ship, much the same as for spent fuel. Ship transport would be employed only if a disposal alternative involving transport across an ocean were implemented. Barge transport would likely be employed only if both the repository and the fuel reprocessing plant were located on or very near navigable waterways. Rail transport would likely be preferred to truck transport because of the greater capacity of the rail casks.

We assume in this Statement that all transport of solidified high-level waste is by rail. Casks for such use have not been constructed but some have been designed (Perona and Blomeke 1972, Peterson and Rhoads 1977). These designs provide for transport of multiple waste canisters in a single cask and incorporate many features of spent fuel cask designs.

The rail cask chosen as the basis for this study is a lead-filled double-walled stainless steel cylinder weighing about 100 MT (220,000 lb) (Peterson and Rhoads 1972). Neutron shielding is furnished by a water jacket that surrounds the cask body. The cask will dissipate up to 50 kW of internally generated heat. High-level waste canisters are held in an aluminum insert that fits into the cask cavity. Different inserts can accommodate nine 0.30-m dia (12-in.), thirteen 0.25-m dia (10-in.), twenty 0.20-m dia, or thirty-six 0.15-m dia (6-in.) waste canisters. Each of these configurations transports the same quantity of waste. Thus, regardless of the canister heat generation limit imposed by disposal constraints, the required number of shipments does not vary.

The cask is transported on a special six-axle rail car. The gross shipping weight of the loaded cask and rail car is about 350 MT (330,000 lb). Casks used for ship transport, in the event this is required by the choice of a disposal alternative, would require adaptation or modification of existing design.

4.5.3 TRU Waste Transport

Transport of TRU wastes is also required in the reprocessing cycle. These wastes are considered here in two categories: 1) fuel residues, which we assume to be packaged in special canisters; and 2) other solid wastes, which we assume to be packaged in steel drums or boxes (except for a small quantity in special canisters). Only truck and rail transport are considered.

4.5.3.1 Fuel Residue Transport

Fuel residues (spent fuel hulls and hardware) are assumed in this Statement to be packaged in special stainless steel canisters (Section 4.3.3.1). Casks for transport of such canisters have not been built, but it is reasonable to assume that the design and construction of such casks present no new problems.

Fuel residue casks may be shipped by rail or truck. Because rail casks could have a greater capacity and because both reprocessing plants and repositories will have rail service, we assume in this Statement that all fuel residue shipments are by rail. For planning purposes a rail cask has been postulated that would transport three canisters. The conceptual cask is a lead-filled, double-walled stainless steel cylinder weighing about 45 MT (140,000 lb). An insert would position the three canisters inside the cask cavity and would act as a heat conduction path from the waste canisters to the inner surface of the cavity wall. Neither cooling fins nor neutron shielding are required.

A truck cask that would transport one fuel residue canister has also been postulated for comparison purposes. This conceptual truck cask is assumed to be a lead-filled, double-walled stainless steel cylinder weighing about 20 MT (43,000 lb).

4.5.3.2 Other TRU Waste Transport

Other TRU wastes to be transported are the packages resulting from the treatment and packaging operations for failed equipment and other miscellaneous TRU wastes (described in Sections 4.3.3.2 through 4.3.3.4). These packages are mainly steel drums and steel boxes, but special canisters like those used for fuel residue are used in this Statement for a portion of the failed equipment. We assume that all of these packages require shipment in casks or overpacks that meet Type B packaging standards, even though it is likely that some could contain a small enough quantity of radioactivity to permit their shipment in Type A packages. Typical Type A packaging includes steel drums, wooden boxes, and steel boxes that prevent loss or dispersal of radioactive contents and retain radioactive shielding if required when subjected to stresses associated with normal transport. Type B packaging must meet these standards, but also must be able to survive a series of hypothetical accident test conditions.

Shipments of these wastes could be made by truck or rail. We assume here that most of these shipments will be by truck. The special canisters containing some of the failed equipment are transported by rail along with the fuel residue waste.

Drums and boxes that have surface dose rates below 200 mR/hr and can be contact-handled are assumed to be transported in a Super Tiger.[®] A Super Tiger is a double-walled steel box with a fire-resistant polyurethane foam filler for shock and thermal insulation. Three pallets, each containing twelve 55-gal drums or three steel boxes (1.2 x 1.2 x 1.8 m), can be accommodated in a Super Tiger. The maximum payload is about 14 MT (30,000 lb), and the empty weight is 6.8 MT (15,000 lb). Super Tigers can be carried by either truck or rail.

[®] Registered Trademark of Protective Packaging, a subsidiary of Nuclear Engineering Company.

Drums that have surface dose rates in the range 200 mR/hr to 1 R/hr require remote handling and are assumed here to be transported in a shielded van that meets Type B package standards or in a Super Tiger-type overpack that incorporates some shielding even though such packages are not currently available or designed. Drums that have surface dose rates in the range 1 to 10 R/hr are assumed here to be transported in casks having an equivalent shield thickness of 5 cm lead + 2 cm steel; a capacity of 14 drums per cask is assumed. Drums with surface dose rates above 10 R/hr are assumed to be transported in casks with an equivalent shield thickness of 10 cm lead + 2.5 cm steel; a capacity of six drums per cask is assumed for planning purposes.

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4.6 DECOMMISSIONING OF RETIRED FACILITIES

Portions of fuel cycle facilities become contaminated with radionuclides during their use. Upon retirement these facilities become a waste that must be managed. Management of this waste is commonly termed decommissioning. Various alternatives are available for decommissioning retired fuel cycle facilities, as discussed in DOE/ET-0028, Section 8.0. Much of this information was extrapolated from results of detailed studies of the technology, safety, and costs of decommissioning nuclear facilities that have been performed at PNL for the NRC (see Schneider and Jenkins 1977, Smith et al. 1978, Smith and Polentz 1978, Jenkins et al. 1979). In this Statement we assume that dismantlement is required and have chosen one of two basic decommissioning modes: either immediate dismantlement, or safe storage with deferred dismantlement.

In immediate dismantlement, all radioactive contamination above regulatory limits is removed from the facility to an approved disposal or storage site shortly after the facility is shut down. Depending on further uses of the site, noncontaminated portions of the facility remaining after dismantlement may be demolished and removed or they may be used for other purposes.

In safe storage with deferred dismantlement, the facility is prepared at shutdown to be left in place for an extended time before it is dismantled. The purpose of this deferment is to allow some of the radionuclides to decay so that radiation exposure during the decommissioning will be reduced. Consideration has been given to both passive safe storage and hardened safe storage methods. These methods differ in the strength and complexity of the barriers installed and in the amount of maintenance and surveillance required during the time of deferment. This time period is termed the continuing care period.

Among the techniques used in decommissioning are chemical decontamination, mechanical decontamination, equipment deactivation and removal, and isolation of contaminated areas. Chemical decontamination is often carried out during the initial stages of a decommissioning operation to reduce radiation levels and remove relatively mobile contamination. Decontamination solutions may include corrosive acids, complexants, detergents, and high-pressure water or steam. These liquids are generally concentrated by evaporation, and the concentrated waste is then immobilized for disposal or storage.

Mechanical decontamination is required to remove residual radioactive contamination from structural surfaces. These activities are minimal when the facility is being prepared for safe storage but are extensive during dismantlement. Contaminated steel structural components or liners may be removed by sectioning in place with plasma torches, arc saws, or explosives. Contaminated concrete can be removed with explosives, by drilling and rock-splitting, or by jackhammering.

Equipment deactivation is done during preparation for safe storage and equipment is removed at the time of dismantlement. Deactivation involves removing bulk quantities of process materials or other hazardous substances, closing valves or installing blank

flanges, and disconnecting electricity and other utilities. Steel equipment can be sectioned (if necessary) and removed using cutting torches, saws, and/or explosive cutting techniques.

Isolation of contaminated areas is required for safe storage. Airtight barriers are constructed around contaminated areas (existing facility structures form most of the barrier) and existing penetrations into contaminated areas are sealed off. HEPA-filtered vents may be installed to accommodate changes in air pressure caused by temperature fluctuations. The barriers constructed for hardened safe storage typically are more substantial and require less maintenance during the continuing care period than the barriers constructed for passive safe storage.

This Statement addresses decommissioning only of the fuel cycle facilities subsequent to the nuclear power plants and decommissioning waste treatment of only the TRU wastes. All of the decommissioning wastes from the example once-through fuel cycle and a portion of those from the reprocessing fuel cycle are expected to be non-TRU wastes.

The fuel cycle facilities examined in detail in this Statement include the away-from-reactor storage facilities (AFRs) in the once-through cycle and fuel reprocessing plants (FRPs) and the mixed-oxide fuel fabrication facilities (MOX-FFPs) in the fuel reprocessing cycle. Interim waste storage facilities other than AFRs also require decommissioning, but this Statement does not consider their decommissioning in detail. Estimates of costs for decommissioning these other waste storage facilities are included in total waste management costs but other effects are too small to make a significant contribution to total impacts.

Immediate dismantlement is the example decommissioning method selected here for the AFR. All of the wastes are expected to be non-TRU waste.

For decommissioning an FRP, we assume a 30-yr period of passive safe storage before dismantlement as the example method. Both TRU and non-TRU wastes are expected to result, but only the TRU portion is considered for disposal here. Most of the combustible and wet wastes generated during the safe storage period are treated with the installed waste treatment equipment, and the packaged wastes are stored in the facility until it is dismantled. The wastes generated near the end of the safe storage period, after the waste treatment facilities have been shut down, are packaged and shipped offsite to a treatment facility before being sent to disposal or storage, as are those wastes generated during the 30-yr continuing care period. The noncombustible wastes generated during dismantlement are packaged without treatment and shipped to disposal or storage.

Because of the low levels of gamma radiation, immediate dismantlement is the decommissioning method assumed here for a MOX-FFP. All of the radioactive wastes resulting from these operations are assumed to be TRU wastes. All wet wastes and most combustible wastes are assumed to be treated with the existing onsite waste treatment equipment. The combustible waste generated after the onsite waste treatment facilities have been shut down is packaged and shipped offsite for treatment prior to disposal or storage. The noncombustible waste and the treated wet and combustible wastes are packaged and shipped to disposal or storage.

Alternative decommissioning methods involving hardened safe storage were also examined for the three facilities. A continuing care period of about 100 years was considered for an AFR, while periods of about 1000 years were considered for the FRP and the MOX-FFP. The 1000-year storage period was used to provide a conservative upper bound to the environmental effects from this activity. A proposed EPA waste storage criterion would limit the safe storage period to about 100 years.

More detail on the wastes resulting from the decommissioning of these facilities is contained in DOE/ET-0028 (Section 8.0 and Section 10--Appendix A). Estimated quantities and radionuclide content of the untreated wastes from the example decommissioning processes are given in Table 4.6.1. The quantities are markedly lower than those presented earlier (Table 4.2.3) for the wastes resulting from operation of these facilities. The radionuclide content is also much lower. Quantities of packaged waste resulting from treatment of the decommissioning wastes are listed in Table 4.6.2.

The radionuclide releases estimated to occur during the decommissioning steps and during the TRU-decommissioning waste treatment operations are presented in Table 4.6.3. Except for the water from the fuel storage basins at an AFR, no release of radioactive liquids is planned. The water from the storage basins at the FRP is vaporized for discharge (using an existing vaporizer), as is the water present in the decontamination solutions.

TABLE 4.6.1. Volumes and Radionuclide Content of TRU Wastes Resulting from Decommissioning of Reprocessing Cycle Facilities

Waste Category	Facility	Volume, m ³ /GWe-yr	Radionuclide Content, Ci/GWe-yr ^(a)						
			Fission Products			Actinides			
			⁹⁰ Sr	¹³⁷ Cs	Total All	²³⁹ Pu	²⁴¹ Pu	²⁴¹ Am	Total All
Noncombustible Waste (Equipment and Structural Material)	FRP	1.4	4.7 x 10 ⁻¹	7.5 x 10 ⁻¹	2.4	7.4 x 10 ⁻³	4.0 x 10 ⁻¹	1.1 x 10 ⁻¹	6.4 x 10 ⁻¹
	MOX-FFP	1.5	-----	-----	-----	2.4 x 10 ⁻¹	6.0 x 10 ¹	1.9	6.5 x 10 ¹
Compactable and Com- bustible Waste									
	Trash								
	FRP	0.15	4.8 x 10 ⁻⁴	7.6 x 10 ⁻⁴	2.4 x 10 ⁻³	8.4 x 10 ⁻⁶	4.6 x 10 ⁻⁴	1.3 x 10 ⁻⁴	7.2 x 10 ⁻⁴
	MOX-FFP	0.06	-----	-----	-----	6.1 x 10 ⁻³	1.5	4.9 x 10 ⁻²	1.7
Filters	FRP	0.25	1.2 x 10 ⁻¹	1.9 x 10 ⁻¹	6.1 x 10 ⁻¹	5.2 x 10 ⁻³	2.8 x 10 ⁻¹	8.0 x 10 ⁻²	4.4 x 10 ⁻¹
	MOX-FFP	0.02	-----	-----	-----	2.2 x 10 ⁻¹	5.6 x 10 ¹	1.8	6.1 x 10 ¹
Concentrated Liquids, Wet Wastes, and Par- ticulate Solids	FRP	0.15	7.9 x 10 ⁻²	1.3 x 10 ⁻¹	4.0 x 10 ⁻¹	1.4 x 10 ⁻³	7.6 x 10 ⁻²	2.2 x 10 ⁻²	1.2 x 10 ⁻¹
	MOX-FFP	0.19	-----	-----	-----	9.0 x 10 ⁻²	2.2 x 10 ¹	7.1 x 10 ⁻¹	2.4 x 10 ¹
Total		3.7	6.7 x 10 ⁻¹	1.1	3.4	5.7 x 10 ⁻¹	1.4 x 10 ²	4.7	1.5 x 10 ²

(a) At the time of assumed dismantlement (30 years after shutdown for the FRP and at the time of shutdown for the MOX-FFP), based on 30 years of facility operation before decommissioning.

TABLE 4.6.2 Estimated Quantities of Packaged TRU-Decommissioning Wastes

<u>Waste Category</u>	<u>Facility</u>	<u>Package Type^(a)</u>	<u>Packages/GWe-yr^(b)</u>
Noncombustible Waste (Equipment and Structural Materials)	FRP	Box	0.028
		Drum (55-gal)	6.0
	MOX-FFP	Box	0.094
		Drum (55-gal)	5.4
HEPA Filters	FRP	Drum (80-gal)	2.2
	MOX-FFP	Drum (80-gal)	0.14
Other	FRP	Drum (55-gal)	1.2
	MOX-FFP	Drum (55-gal)	0.63

(a) All packages are anticipated to have surface dose rates below 200 mR/hr, and can thus be contact-handled.

(b) Based on 30 years of facility operation before decommissioning.

TABLE 4.6.3. Radionuclides Released on Example Decommissioning of Facilities

Fission Products	Radionuclide Release ^(a) at FRP, Ci			Radionuclide Release ^(a) at MOX-FFP, Ci		Radionuclide Release at AFR, Ci	
	Safe Storage	Dismantlement	TRU Waste Treatment ^(b)	Dismantlement	TRU Waste Treatment	To Water Bodies	To Atmosphere ^(a)
⁹⁰ Sr	8.0 x 10 ⁻⁴	2.5 x 10 ⁻⁴	7.8 x 10 ⁻¹⁰	----	----	3.6 x 10 ⁻³	7.2 x 10 ⁻⁹
¹⁰⁶ Ru	1.6 x 10 ⁻⁴	----	1.6 x 10 ⁻¹⁰	----	----	8.0 x 10 ⁻⁶	1.6 x 10 ⁻¹¹
¹²⁹ I	6.3 x 10 ⁻¹¹	4.2 x 10 ⁻¹¹	6.3 x 10 ⁻¹⁷	----	----	----	----
¹³⁴ Cs	1.3 x 10 ⁻³	5.6 x 10 ⁻⁹	2.1 x 10 ⁻¹⁰	----	----	2.1 x 10 ⁻²	4.1 x 10 ⁻⁸
¹³⁷ Cs	2.3 x 10 ⁻³	4.0 x 10 ⁻⁴	1.2 x 10 ⁻⁹	----	----	2.2 x 10 ⁻¹	4.3 x 10 ⁻⁷
¹⁴⁴ Ce	1.7 x 10 ⁻⁴	-----	1.6 x 10 ⁻¹⁰	----	----	1.5 x 10 ⁻⁵	3.0 x 10 ⁻¹¹
Total All Fission Products	7.3 x 10 ⁻³	1.3 x 10 ⁻³	5.1 x 10 ⁻⁹	----	----	2.4 x 10 ⁻¹	4.7 x 10 ⁻⁷
<u>Actinides</u>							
²³⁸ Pu	3.0 x 10 ⁻⁵	2.4 x 10 ⁻⁸	9.3 x 10 ⁻¹¹	1.2 x 10 ⁻⁵	4.2 x 10 ⁻¹¹	----	----
²³⁹ Pu	2.2 x 10 ⁻⁶	2.2 x 10 ⁻⁹	6.8 x 10 ⁻¹²	8.8 x 10 ⁻⁷	3.1 x 10 ⁻¹²	----	----
²⁴⁰ Pu	4.4 x 10 ⁻⁶	4.5 x 10 ⁻⁹	1.4 x 10 ⁻¹¹	1.8 x 10 ⁻⁶	6.3 x 10 ⁻¹²	----	----
²⁴¹ Pu	5.6 x 10 ⁻⁴	1.2 x 10 ⁻⁷	1.7 x 10 ⁻⁹	2.2 x 10 ⁻⁴	7.6 x 10 ⁻¹⁰	----	----
²⁴¹ Am	2.0 x 10 ⁻⁵	3.4 x 10 ⁻⁸	6.2 x 10 ⁻¹¹	7.0 x 10 ⁻⁶	2.4 x 10 ⁻¹¹	----	----
²⁴² Cm	1.5 x 10 ⁻⁶	1.9 x 10 ⁻¹⁰	4.6 x 10 ⁻¹²	----	----	----	----
²⁴⁴ Cm	2.6 x 10 ⁻⁵	7.2 x 10 ⁻⁹	8.1 x 10 ⁻¹¹	----	----	----	----
Total All Actinides	6.5 x 10 ⁻⁴	1.9 x 10 ⁻⁷	2.0 x 10 ⁻⁹	2.4 x 10 ⁻⁴	8.4 x 10 ⁻¹⁰	----	----
<u>Activation Products</u>							
⁵⁵ Fe	2.3 x 10 ⁻⁴	----	----	----	----	6.5 x 10 ⁻³	1.3 x 10 ⁻⁸
⁶⁰ Co	6.5 x 10 ⁻⁵	----	----	----	----	9.5 x 10 ⁻³	1.9 x 10 ⁻⁸
Total All Activation Products	6.5 x 10 ⁻⁴	----	----	----	----	1.7 x 10 ⁻²	3.3 x 10 ⁻⁸

(a) Released from the facility exhaust stack.

(b) Based on the radionuclide content at the time of shutdown.

REFERENCES FOR SECTION 4.6

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- Smith, R. I., G. J. Konzek, and W. E. Kennedy, Jr. 1978. Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station. NUREG/CR-0130, U.S. Nuclear Regulatory Commission, Washington, D.C.
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4.7 ENVIRONMENTAL IMPACTS OF PREDISPOSAL OPERATIONS

Impacts of predisposal operations, including construction and decommissioning of waste management facilities and transport casks, operation of waste management facilities, and transportation of spent fuel and reprocessing wastes, are described here. Impacts considered include land, water and resource use, socioeconomic impacts, and radiological effects. The sources of this information are DOE/ET-0028 and DOE/ET-0029, which may be consulted for details.

The operational impacts discussed here are based on routine operations. Accidents and their impacts are discussed in Section 4.8. Source terms for routine releases of radioactive effluents do, however, include releases from minor accidents at reference facilities.

4.7.1 Environmental Impacts Related to Predisposal Operations for the Once-Through Fuel Cycle

The predisposal operations in the example once-through fuel cycle of this Statement include: 1) initial storage of unpackaged spent fuel in water basins either at the reactors or in away-from-reactor storage facilities (AFRs), 2) transportation of spent fuel to the disposal site (and between storage sites, if necessary), and 3) packaging of the spent fuel. An additional operation, extended storage of packaged spent fuel, is also evaluated for possible use in case there is a long delay in repository availability. The impacts of constructing, operating, and decommissioning these facilities are covered in this section.

The impacts of the fuel packaging facilities are included with those of the AFRs in this section, as in DOE/ET-0029, even though the example case for this Statement assumes that the fuel packaging facilities are located at the disposal sites. Fuel packaging facilities might also be located at the extended storage facilities, if such storage is implemented. The fuel packaging facility impacts would be essentially the same at any of the three locations.

These predisposal operations assume that the spent fuel will be disposed in a mined geologic repository within the continental U.S. The use of alternative disposal concepts could alter the number and type of predisposal facilities required. The use of a concept involving disposal outside the continental U.S. (i.e., island, subseabed, or ice sheet disposal) requires the use of additional transportation facilities (i.e., ships and docking facilities) and possible additional storage facilities. Use of the space disposal, rock melting, or well injection concepts requires the use of processing plants to obtain suitable waste forms. Impacts of such processing plants would be similar to those of a fuel reprocessing plant in the reprocessing cycle case.

4.7.1.1 Resource Commitments for Once-Through Fuel Cycle Waste Management

Land use commitments for a 3000 MTHM AFR with a fuel packaging facility are about 40 ha, of which 14 ha will be cleared for construction.

Water use will be $6 \times 10^4 \text{ m}^3$ during construction and $2.5 \times 10^5 \text{ m}^3$ per year during operation. As long as water can be supplied from rivers such as the reference R River (Appendix F), water use should represent a small fraction (~ 0.001) of the average river flow, and no significant impact will result from its withdrawal. Site selection should avoid adverse effects on aquatic systems and other downstream uses of water.

Other resource commitments during construction and operation of an AFR are presented in Table 4.7.1. Resource commitments for fabrication and use of spent fuel shipping casks are presented in Table 4.7.2.

Resource commitments during decommissioning consist mainly of steel, electricity, and diesel fuel. Total commitments of these resources during decommissioning will be small fractions of construction commitments.

TABLE 4.7.1. Resource Commitments for Construction and Operation of an Example AFR

	<u>Construction</u>	<u>Operation(a)</u>
<u>Materials</u>		
Concrete, m^3	2.3×10^4	---
Steel, MT	1.1×10^4	---
Stainless Steel, MT	6.1×10^3	---
Copper, MT	2.7×10^1	---
Lumber, m^3	1.3×10^3	---
<u>Energy</u>		
Propane, m^3	5.7×10^2	---
Diesel Fuel, m^3	5.7×10^3	---
Gasoline, m^3	3.8×10^3	---
Electricity, kWh	2.8×10^6	7.8×10^8
Manpower, man-yr	2.5×10^3	2.4×10^3

(a) Based on operation for 30 years.

TABLE 4.7.2. Resource Commitments for Fabrication and Use of Spent Fuel Shipping Casks(a)

<u>Resource</u>	<u>MT/Cask</u>	<u>(m^3/km) per Shipment</u>
Stainless Steel	26	--
Lead	65	--
Depleted Uranium	5	--
Diesel Fuel	--	0.0016

(a) For an "average" cask for train transport of spent fuel, which has a spent fuel capacity of about 4 MTHM.

4.7.1.2 Nonradiological Effluents of Once-Through Fuel Cycle Waste Management

Nonradiological effluents from AFR construction include dust and pollutants from machinery operation. Burning the quantities of fossil fuels listed in Table 4.7.1 also results in air pollution emissions, but concentrations in air at the fence line from construction and operation are not expected to degrade air quality beyond applicable limits (40 CFR 50).

The major nonradiological effluent from operation of an AFR is the release of about 5×10^8 MJ/yr of heat through the cooling tower. These thermal releases are not expected to have any significant effects, nor any measurable micrometeorological effects. Predicted nonradiological effluent air concentrations from AFR operations will be considerably below applicable Federal air quality standards or naturally occurring gaseous concentrations.

Nonradiological effluents from decommissioning will be comparable to effluents during construction of the AFR and are not expected to result in any degradation of air quality.

4.7.1.3 Radiological Effects of Once-Through Fuel Cycle Waste Management

During planned operation of an AFR, the only exposure pathway to man is via airborne effluents; there are no planned releases of radioactivity to ground or water. During decommissioning, it is assumed that the purified pool water and the contained radionuclides are released to the local water bodies, however. A summary of the 70-year total body doses to the work force and the regional population during operation and decommissioning of an example AFR is given in Table 4.7.3.

In this Statement, 100 to 800 health effects are postulated to result in the exposed population per million man-rem. Based on calculated doses to the work force, 0 to 3 health effects are expected over a 70-year period as a result of operation of one 3000 MTHM AFR.

The regional population dose estimated here is a few hundred times lower than that estimated elsewhere for similar facilities (DOE/EIS-0015, Appendix B). This difference results mainly from the extra conservatism used in the other study. Both studies indicate that the doses to the regional population expected to result from AFR operation are very small in comparison to the doses to the same people during the same time period from naturally occurring sources.

TABLE 4.7.3. Doses Resulting From Operation and Decommissioning of an AFR

	70-Year Whole-Body Dose, man-rem	
	Operation	Decommissioning
Regional Population	1.4 ^(a)	9.8×10^1 ^(a)
Work Force	3.6×10^3	7.0×10^1

(a) The dose to the population from naturally occurring sources during the same period is about 1×10^7 man-rem.

No significant releases of radioactive material are expected during transportation of spent fuel under normal operating circumstances. However, members of the transport work force and of the population along the shipping route will receive dose from the direct radiation from the shipments. The dose for each 4 MTHM rail shipment is estimated to be 7.8×10^{-6} man-rem/km to the regional population and 5×10^{-6} man-rem/km to the transport work force. For each 0.4 MTHM truck shipment, the doses are estimated to be 2.2×10^{-6} man-rem/km to the regional population and 5×10^{-5} man-rem/km to the transport work force. For a 1,600-km shipment distance, the dose to the population for a rail shipment is 0.012 man-rem/shipment. For comparison, the estimated dose to the same population from naturally occurring sources is 230 man-rem/day.

4.7.1.4 Ecological Effects of Once-Through Fuel Cycle Waste Management

Construction of an example AFR will remove about 10 ha from its present assumed use for agriculture and wildlife for the life of the plant. While this change in land use will reduce its utility as habitat for wildlife, no significant ecological impacts to the region are expected. Disturbance of animals from fugitive dust, noise, and human activities during construction will be confined mainly to the 405-ha AFR restricted area. Erosion from run-off may deposit silt in nearby surface waters unless drainage is controlled by proper ditching, grading, and silt catchment. After construction is completed and vegetation is reestablished or surfacing is completed in the disturbed areas, the erosion problem will be reduced or eliminated.

The maximum concentrations of airborne particulates, sulfur dioxide, and carbon monoxide will occur within the 405-ha AFR restricted area. Particulate concentrations at the site during construction and decommissioning are estimated to be within Federal ambient air standards. Levels of carbon monoxide and hydrocarbons calculated to be found are only a small fraction of the existing rural air concentrations near the reference site. Concentrations of the other materials are less than applicable standards. Consequently, no measurable detrimental effects on the terrestrial ecosystem are anticipated.

During operation of the AFR, the release of about 5×10^8 MJ/yr of waste heat is not expected to have any ecological impact. No significant effects are expected as a result of discharging the cooling tower blowdown to the local water bodies.

Particulates and gases released to the atmosphere from combustion of fossil fuels during normal transport operation are not expected to be of ecological significance.

4.7.1.5 Socioeconomic Impacts of Once-Through Fuel Cycle Waste Management

Socioeconomic impacts associated with construction and operation of an away-from-reactor storage facility depend largely on the number of persons who move into the county in which the facility will be located. Because of this, estimates were made of the size of the local population influx and their needs for locally provided social services.

The expected socioeconomic impacts of an AFR on reference sites located in the Southeast and Midwest U.S. are judged to be insignificant. The total number of estimated new in-migrants equals only about 1% of the existing population in both the construction and operation phases. In addition, there are no very large transitions over time and the expected number of in-migrants increases steadily over the life of the project.

The effect of the project is substantially different in the reference Southwest site. The number of in-migrants estimated amounts to about 9% of the existing population during construction and about 6% during operation. This decline in population influx from construction to operations of about one-third sets the stage for a boom and bust type of effect in the Southwest site.

Translating estimated project-related in-migration into socioeconomic impacts is complex and imprecise. Estimates of the level of demand that will be placed on the community to provide social services to the new workers and their families were made by applying a set of factors (Appendix G) to the project in-migration values. The product of these factors indicates how many units of each social service would be "expected" by the in-migrants. The significance of the impacts is primarily related to the capacity of the site county to meet these expectations. The calculated level of expected social services at the three reference sites is given for the year 2000 in Table 4.7.4.

TABLE 4.7.4. Selected Social Service Demands Associated with In-Migration Related to a 3000 MTHM AFR

	Expected Demand in the Year 2000		
	<u>Southeast Site</u>	<u>Midwest Site</u>	<u>Southwest Site</u>
Health			
Physicians	0	1	3
Nurses	1	3	9
Dentists	0	0	1
Hospital beds	1	3	11
Nursing care beds	1	3	7
Education			
Teachers	4	7	43
Classroom space, m ² (9-12)	480	960	5180
Sanitation, m ³ /day			
Water treatment	170	320	1840
Liquid waste	110	210	1260
Safety			
Firemen	0	0	2
Policemen	1	1	7
Recreation, ha			
Neighborhood parks	0	1	3
Government			
Administrative staff	0	1	3

4.7.2 Environmental Impacts Related to Predisposal Operations for the Reprocessing Fuel Cycle

Waste treatment operations required in the reprocessing fuel cycle were discussed in Sections 4.3.2 through 4.3.5 for fuel reprocessing plants (FRPs) and mixed-oxide fuel fabrication plants (MOX-FFPs). Potential waste storage requirements were discussed in Sections 4.4.2 through 4.4.4. In this section we will summarize the environmental effects of these waste management operations. The effects will be summarized for three different reference facilities: 1) a 2000 MTHM/yr FRP, 2) a 400 MTHM/yr MOX-FFP, and 3) a retrievable waste storage facility (RWSF) that has capacity to store all the high-level and TRU wastes from FRPs and MOX-FFPs during the passage of 45,000 MTHM through the fuel cycle. An RWSF will be necessary only if reprocessing is initiated significantly before a repository is available.

The environmental effects of waste treatment, storage, and transportation are summarized here for the example concepts defined in Sections 4.3, 4.4 and 4.5 for the reprocessing fuel cycle. The environmental effects of alternative concepts were also examined in DOE/ET-0029; only in the off-gas case, where the results are significantly different from those of the example concepts, are the alternatives discussed here.

The use of other than deep geologic repositories for disposal of the high-level waste could alter the number and type of waste management facilities required. As in the once-through cycle, additional transportation facilities such as ships and docking facilities would be required for disposal by the island, subseabed, or ice sheet disposal concepts. Use of the rock melting or well injection concepts to dispose of liquid waste would eliminate the need for high-level waste solidification and solidified high-level waste storage facilities but would probably require the addition of substantial liquid high-level waste storage facilities. Use of the space disposal concept would require additional chemical processing facilities and, perhaps, the addition of substantial liquid high-level waste storage facilities.

4.7.2.1 Resource Commitments in Reprocessing Fuel Cycle Waste Management

Land use commitments for waste management facilities at the reference FRP are about 19 ha compared to 60 ha for the production facilities. At the reference MOX-FFP, the waste management facilities occupy about 0.3 ha of the 6 ha required for the production facilities. An RWSF of the reference size would require 170 ha for buildings and storage areas.

Water used during construction of waste management facilities amounts to about $1.4 \times 10^5 \text{ m}^3$, $5.9 \times 10^3 \text{ m}^3$ and $3.1 \times 10^5 \text{ m}^3$, for the FRP, MOX-FFP, and RWSF, respectively. If these quantities of water are withdrawn over the period of construction from a river such as R River, as described in the reference environment, the impact on downstream uses will be insignificant.

Resources committed for construction and operation of the waste management facilities are summarized in Table 4.7.5. Resources for construction and use of waste shipping

TABLE 4.7.5. Resource Commitments for Construction and Operation of Reprocessing Fuel Cycle Waste Management Facilities

	Waste Mgmt. Facilities at Example FRP		Waste Mgmt. Facilities at Example MOX-FFP		Example RWSF	
	Construction	Operation ^(a)	Construction	Operation ^(a)	Construction	Operation ^(a)
Material						
Concrete, m ³	7.8 x 10 ⁴		3.0 x 10 ³		2.6 x 10 ⁵	
Cement, MT		3.3 x 10 ⁴		1.1 x 10 ⁴		2.2 x 10 ⁴
Steel, MT	1.8 x 10 ⁴		6.6 x 10 ²		5.5 x 10 ⁴	1.1 x 10 ⁵
Stainless Steel, MT		6.6 x 10 ³				
Copper, MT	2.0 x 10 ²		6.9		3.0 x 10 ²	
Lumber, m ³	5.1 x 10 ³		1.8 x 10 ²		1.3 x 10 ⁴	
Plywood, m ²		1.0 x 10 ⁵				3.0 x 10 ⁵
Energy and Utilities						
Propane, m ³	1.3 x 10 ³	8.4 x 10 ⁶	4.1 x 10 ¹	3.0 x 10 ⁵	3.5 x 10 ³	
Diesel Fuel, m ³	1.2 x 10 ⁴	7.2 x 10 ²	4.2 x 10 ²	1.6 x 10 ³	3.5 x 10 ⁴	2.2 x 10 ⁴
Gasoline, m ³	8.7 x 10 ³		3.2 x 10 ²		2.5 x 10 ⁴	
Electricity, kWh	6.4 x 10 ⁶	2.7 x 10 ⁹	2.8 x 10 ⁵	4.2 x 10 ⁷	1.7 x 10 ⁴	1.4 x 10 ⁹
Water consumed, m ³	1.4 x 10 ⁵	1.3 x 10 ⁷	5.9 x 10 ³	2.5 x 10 ⁴	3.1 x 10 ⁵	9.0 x 10 ³
Manpower, man-yr	4.0 x 10 ³	4.5 x 10 ³	1.9 x 10 ²	2.6 x 10 ²	5.1 x 10 ³	2.6 x 10 ³

(a) Based on operation for 30 years.

containers are given in Table 4.7.6. These resource commitments are small in comparison with those of the FRP and MOX-FFP production facilities and in an absolute sense are not expected to have a significant impact on available supplies of these materials or energy sources. Energy and materials required for decommissioning do not add significantly to the quantities of resources required for construction.

4.7.2.2 Nonradiological Effluents of Reprocessing Fuel Cycle Waste Management

Nonradioactive pollutants released to the atmosphere during construction of the FRP and MOX-FFP waste management facilities and the RWSF result from the combustion of fuel in construction vehicles and machinery, fugitive dust from ground-clearing operations, and particulates from concrete batch operations.

Offsite concentrations of carbon monoxide, hydrocarbons, and particulates resulting from construction force traffic and construction equipment emissions are projected to be less than Federal ambient air quality standards. (Onsite concentrations of particulates at the FRP and MOX-FFP construction sites were found to exceed the air quality standards; this will occur primarily as a result of construction of FRP and MOX-FFP production facilities and is a normal situation at sites of heavy construction.) Evaluation of sulfur dioxide and nitrogen oxide emissions indicates no significant effects.

The release of about 1×10^9 MJ of waste heat per year from the example FRP waste management facilities is comparable to the release of heat from a small city or town (30,000 persons) and is not expected to produce any significant effect on the environment.

Predicted concentrations of pollutants in air from waste management operations will be a small fraction of Federal air quality standards, threshold limit value concentrations

TABLE 4.7.6. Resource Commitments for Construction and Use of Waste Shipping Containers

<u>Shipping Container</u>	<u>Example Capacity</u>	<u>Material Used in Construction, MT/cask</u>		<u>Diesel Fuel Used per Shipment, m³/km</u>
		<u>Stainless Steel</u>	<u>Lead</u>	
High-level waste cask	Solidified HLW from 27 MTHM	25	75	0.0020
Fuel residue cask	3 fuel residue canisters (residue from 12 MTHM)	16	49	0.0013
6-drum cask	Six 55-gal drums	4	15	
14-drum cask	Fourteen 55-gal drums	5	14	
Shielded overpack	Thirty-six 55-gal drums	7	12	
Unshielded overpack	Thirty-six 55-gal drums (or equivalent volume of boxes)	7	0	0.0010

(those to which nearly all workers may be repeatedly exposed without adverse effect), and naturally occurring gaseous concentrations. Consequently, no detrimental effects are anticipated.

Water withdrawn from the R River for waste management facility operation is not expected to have adverse effects on local water supplies.

4.7.2.3 Radiological Effects of Reprocessing Fuel Cycle Waste Management

During planned operation of the waste management facilities, the only exposure pathway to man is via airborne effluents; there are no planned releases to the ground or water. For transportation of radioactive wastes under normal circumstances, no radioactive materials will be released via any pathway. However, individuals will receive doses from the direct radiation from passing rail and truck shipments.

A summary of the 70-year whole-body doses to the regional population for the individual waste management activities at the example facilities is given in Table 4.7.7.

Ninety percent of the 70-year whole-body dose to the regional population from waste management operations results from releases from the off-gas system at the FRP. The example system, which partially collects volatilized ruthenium, iodine, carbon and krypton, results in a 70-year whole-body dose to the regional population of 8300 man-rem. Should carbon and krypton be totally released, the dose would be increased to 9900 man-rem, while no treatment, i.e., release of volatilized ruthenium, iodine, carbon and krypton would increase the whole-body dose to 1.6×10^4 man-rem and result in a thyroid dose of 1×10^6 man-rem. The annual thyroid dose to the maximum individual from FRP off-gas effluents without treatment would be 0.16 rem compared to 0.002 rem with treatment. Use of the example system provides reasonable assurance that ^{85}Kr and ^{129}I releases per gigawatt-year will be within limits specified in 40 CFR 190.

The example krypton collection and storage system reduces the worldwide 70-year total body dose due to ^{85}Kr from 2.4×10^5 man-rem to 3.6×10^4 man-rem per FRP. Thus 2.0×10^5 man-rem of exposure is saved by concentrating and storing krypton. The present worth dollar cost of this savings is estimated to be \$230 million; the cost per man-rem saved is thus approximately \$1200. If krypton were totally released during reprocessing, the number of health effects expected to result from the ^{85}Kr radiation would be 24 to 190 per FRP. Implementation of the example krypton collection and storage system would reduce the expected number of health effects to 4 to 29 per FRP. This reduction of from 20 to 160 health effects may be compared to an estimated 60 disabling injuries and about 1 death per FRP resulting from construction of the krypton collection and storage facilities.

The 70-year whole-body dose to the worldwide population for the example treatment processes at one FRP and one MOX-FFP is 2×10^5 man-rem, which is less than 10^{-5} of the dose due to naturally occurring sources during the same 70-year period.

No significant releases of radioactive material are expected during transportation of the packaged wastes under normal operating circumstances. However, members of the transport work force and of the population along the shipping route will receive dose from the direct radiation from the shipments. These doses to the regional population are estimated to be

TABLE 4.7.7. Dose to Regional Population Due to Operation of an FRP and a MOX-FFP

	<u>70-Year Whole-Body Dose, man-rem (a)</u>
<u>High-Level Wastes</u>	
<u>Treatment</u> --vitrification and encapsulation	8.6×10^2
<u>Storage</u> --water basin	1.2×10^{-2}
<u>TRU Wastes</u>	
<u>Treatment</u>	
Fuel residue--package without compaction	3.5×10^{-5}
Failed equipment and noncombustible waste--package after decontamination and disassembly of failed equipment as required.	.
FRP	6.5×10^{-3}
MOX-FFP	1.2×10^{-3}
Combustible and compactable waste--incineration	
FRP contact-handled	3.3×10^{-10}
FRP remotely handled	2.8
MOX-FFP	1.6×10^{-8}
Wet wastes and particulate solids--cementation	
FRP	1.1×10^{-2}
MOX-FFP	1.7×10^{-4}
<u>Storage</u>	
Fuel residue--dry well	0
Other remotely handled--vault	0
Contact-handled--outdoor surface	0
<u>Gaseous and Airborne Wastes</u>	
<u>Treatment</u>	
FRP--filter and remove Ru, I, C, and Kr	8.3×10^3
MOX-FFP--filter	2.4×10^{-5}
<u>Storage</u>	
Krypton at FRP site(b)	4.0×10^1
TOTAL	9.2×10^3

- (a) The whole-body dose received by the same population over the 70-year commitment period due to naturally occurring sources is 1×10^7 man-rem.
- (b) The dose due to operation of the krypton storage facility is an 80-year commitment which includes 30 years of collection plus 50 years of retention before release.

3.7×10^{-6} man-rem/km per shipment of solidified HLW or fuel residue and 1.1×10^{-6} man-rem/km per shipment of other TRU wastes. The doses to the transport work force are estimated to be 5×10^{-6} man-rem/km per shipment of solidified HLW or fuel residue and 5×10^{-5} man-rem/km per shipment of other TRU wastes. Shipments of HLW and fuel residue are assumed to be by rail and those of the other TRU wastes are assumed to be by truck.

Table 4.7.8 presents additional 70-year whole-body dose data. Included here are estimates of the doses to the work force as well as to the regional population and also the doses during transportation of the high-level and TRU wastes generated during the lifetimes of the facilities.

Doses to the work force and the regional population during decommissioning will be 10% of the 70-year total body dose resulting from operation of the facilities, assuming a safe storage period of 30 years before dismantlement of the FRP.

TABLE 4.7.8. Example Reprocessing Cycle Waste Management Operations at Individual Facilities^(a)

	70-Year Whole-Body Dose (man-rem) to:	
	Work Force	Regional Population ^(b)
FRP Waste Management Facilities	14,000	9,200
MOX-FFP Waste Management Facilities	2,700	0.0014
RWSF	3,600	0.001
Waste Transportation	<u>7,200</u>	<u>140</u>
	27,500	9,300

(a) 30-year operation in each case.

(b) The dose to the regional population from naturally occurring sources is about 1×10^7 man-rem.

In this Statement, 100 to 800 health effects are postulated to occur in the exposed population per million man-rem (see Appendix E). On that basis, the 70-year total body doses to the regional population and the work force listed in Table 4.7.8, suggest that the number of health effects expected to occur as a result of waste management operations at one FRP and one MOX-FFP (plus transportation of wastes to the disposal facility) would be 2 to 20 health effects to the work force and 1 to 8 health effects to the regional population. On this same basis, the regional population dose of 10 million man-rem received from naturally occurring sources over the same 70 years suggests that 1,000 to 8,000 health effects would occur from these naturally occurring sources.

4.7.2.4 Ecological Effects of Reprocessing Fuel Cycle Waste Management

Construction of waste management facilities will remove, for the life of the plants, about 19 ha from its present use for agriculture and wildlife at the reference FRP site, and about 0.3 ha at the reference MOX-FFP site. While this change in land use will eliminate its utility as habitat for wildlife, no significant ecological impacts to the regions as a whole are expected. Disturbance of animals from fugitive dust, noise, and human activities during construction will be confined mainly to the restricted areas (2400 ha for the FRP and 400 ha for the MOX). Erosion caused by run-off may deposit silt in nearby surface waters unless drainage is controlled by proper ditching, grading, and silt catchment. After construction is completed and vegetation is reestablished or surfacing is completed in the disturbed areas, this erosion problem will be reduced.

Calculated carbon monoxide and hydrocarbon levels caused by construction of the waste management facilities are only a small fraction of the existing rural air concentrations near the reference sites. Particulate concentrations are estimated to exceed Federal ambient air standards only on the construction site. Concentrations of the other materials are below acceptable standards. Consequently, no measurable detrimental effects on the offsite terrestrial ecosystem are anticipated.

The release of heat during operation of the waste management facilities is expected to have no ecological impact. No perceptible impacts to the river ecosystem are foreseen from

discharges of cooling tower blowdown. With proper intake structure design and placement in the river, the loss of aquatic organisms through intake screen impingement and entrainment in the cooling water is expected to have no significant impact on the river ecosystem.

Since the concentration of air pollutants resulting from operation of the waste management facilities is several orders of magnitude lower than those allowed by the air quality standards, no impacts to the terrestrial ecosystem are expected. No toxic effects to native plant species in the environment are expected during the life of the facilities or during decommissioning.

Some particulates and gases will be released to the atmosphere from combustion of fossil fuels during normal transport operations; however, these releases are expected to be of no ecological significance.

4.7.2.5 Socioeconomic Impacts of Reprocessing Fuel Cycle Waste Management

Socioeconomic impacts associated with waste management facilities depend largely on the numbers of persons who move into the county in which the facilities will be located. To analyze socioeconomic impacts, therefore, the size of the population influx and the needs for local social services were estimated.

The number of in-migrants resulting from construction and operation of waste management facilities is estimated to be large enough to have a significant socioeconomic impact only in the reference Southwest location for the FRP waste management facilities and the RWSF. In these two cases, the number of in-migrants amounts to about 8% of the existing population during construction and about 4% during operation. These facilities at the reference Southeast and Midwest sites are estimated to give population increases of 1% or less. The MOX-FFP waste management facilities are estimated to give population increases of 0.1% or less at each of the three reference sites.

The translation of estimated project-related in-migration into socioeconomic impacts is complex and imprecise. Estimates of the level of demand that will be placed on the community to provide social services to the new workers and their families were made by applying a set of factors (Appendix G) to the project in-migration values. The product of these factors indicates how many units of each social service would be "expected" by the in-migrants. The severity or significance of these impacts is primarily related to the capacity of the site county to meet these expectations. The calculated level of expected social services at the three sites in different areas of the U.S. is given for the year 2000 in Table 4.7.9.

The most significant demands arise for the Southwest site where an adequate labor pool is not expected to exist. However, the social service demands are small compared to those for the FRP and MOX-FFP production facilities.

TABLE 4.7.9. Selected Social Service Demands Associated with In-Migration Related to Waste Management Facilities at an FRP, a MOX-FFP, and an RWSF

	Expected Demand in the Year 2000								
	Southwest Site			Midwest Site			Southwest Site		
	FRP	MOX-FFP	RWSF	FRP	MOX-FFP	RWSF	FRP	MOX-FFP	RWSF
<u>Personnel</u>									
Physicians,									
Nurses, Dentists	1	0	1	4	0	2	10	0	8
Teachers	3	0	2	6	0	4	37	1	28
Firemen,									
Policemen	1	0	0	1	0	1	8	0	6
Gov't Admin.	0	0	0	1	0	0	3	0	2
<u>Services</u>									
Water Treatment, m ³ /day	150	7	100	290	17	180	1620	23	1250
Liquid Waste, m ³ /day	100	4	70	190	11	120	1080	15	840
<u>Facilities</u>									
Hospital and Nursing Beds	2	0	1	6	0	4	16	0	12
Classroom space, m ² (9-12)	420	20	270	880	50	530	4480	70	3390
Neighborhood Parks, ha	0	0	0	0	0	0	1	0	2

REFERENCES FOR SECTION 4.7

- U.S. Department of Energy. 1979. Technology for Commercial Radioactive Waste Management, DOE/ET-0028, Washington, D.C.
- U.S. Department of Energy. 1979. Environmental Aspects of Commercial Radioactive Waste Management, DOE/ET-0029, Washington, D.C.
- U.S. Department of Energy. 1980. Final Environmental Impact Statement, U.S. Spent Fuel Policy, DOE/EIS-0015, Washington, D.C.

4.8 ACCIDENT IMPACTS FOR PREDISPOSAL OPERATIONS

The environmental impacts of accidents that occur during operation of predisposal systems for both the once-through cycle and for the reprocessing cycle are described in this section. Potential accidents for the predisposal functions of treatment and/or packaging, transport, and storage are discussed here for both cycles.

The environmental impacts of accidents described in this section are representative of impacts from all postulated predisposal accidents. Using a methodology of accident identification and classification that included an umbrella source term, we selected the largest source term in classified release categories for environmental impact analysis. Results of this analysis are summarized here. Umbrella source terms are a conservative representation of releases that result from other accidents in their release category. A description of the methodology used to develop and select umbrella source terms for impact analysis is given in Section 3.2.7. Unless specified otherwise, the maximum-exposed individual in the following discussion is considered to be a member of the general public, not a radiation worker. Accident impacts are generally greater to the public than to the workers.

4.8.1 Accident Impacts for the Once-Through Cycle

This section describes the impacts of postulated accidents for handling spent fuel until it is placed in the disposal facility. Operational and long-term accident impacts from spent fuel disposal are discussed in Sections 5.5 and 5.6.

While extended storage of packaged spent fuel is not included in the example case, it may be desired if the operation of the disposal facility is delayed longer than is now expected. Therefore, analysis of accident impacts of packaged spent fuel storage are included as a contingency.

4.8.1.1 Radiological Impacts from Spent Fuel Transportation Accidents

Safety during transport of radioactive material depends primarily on shipping containers. Shipping containers must meet standards established by the Department of Transportation and the Nuclear Regulatory Commission. Containers holding significant amounts of radioactive material must prevent loss or dispersal of radioactive contents, retain shielding efficiency, ensure nuclear criticality safety, and provide adequate heat dissipation under normal conditions of transport and under specified (hypothetical) accident damage test conditions (49 CFR 173.398). Improbable accidents that exceed the severity of hypothetical tests, accidents caused by equipment failures and accidents that are less severe than the test conditions were considered in this analysis to demonstrate the range of potential occurrences in a transportation environment. Impacts of these accidents are summarized below.

Recent regulations for the shipment of spent fuel require that all shipments of spent fuel be escorted in transit; while severe accidents involving this material are still possible, the chances of occurrence will be reduced with this required increased surveillance. Chances of a period of no action by emergency response personnel following an accident,

which is postulated to result in large releases of radioactive material, may be substantially reduced with these additional transportation personnel. Thus, if a severe accident does occur, consequences may be partially mitigated compared to the severe accidents described here.

Truck and rail transport of spent fuel are both expected to be used in the once-through fuel cycle. Descriptions of the systems considered in the analysis along with detailed accident descriptions are reported in DOE/ET-0028. Dose calculations for postulated accidents are reported in DOE/ET-0029. Accident frequency estimates cited in this section are based on an assumed 250 GWe nuclear industry.

The impacts examined in DOE/ET-0028 and DOE/ET-0029 were developed assuming unpackaged short-cooled (6 months out of the reactor) spent fuel. These impacts are thus much more severe than those from accidents involving long-cooled fuel. They also do not take into account the mitigation of impact that is likely to result from the new escorting regulations.

Similar accidents are also plausible for packaged spent fuel if transportation is required following packaging. However, since packaging provides an additional barrier to release of nuclides in transportation of spent fuel, the releases would be smaller and more infrequent than for unpackaged spent fuel. For this reason, specific accidents for packaged spent fuel transport are not discussed but can be assumed to cause lesser impact than unpackaged spent fuel transport.

Six accidents for truck transport of spent fuel were analyzed: three minor, two moderate, and one severe. The minor accidents involved rollovers, collisions and the undetected leakage of coolant. Only coolant leakage was expected to release radioactive material and could result in a 70-yr accumulated dose to the maximum-exposed individual of 3×10^{-6} rem at an expected frequency of approximately twice per year.

The moderate accident giving the largest release of radioactive material is a fire that activated a pressure relief valve on the cask. A 70-yr accumulated dose of 8×10^{-5} rem to the maximum-exposed individual would occur at an estimated frequency of about once every 50 years.

The severe accident culminating in a long-lasting fire results in a 70-yr accumulated dose to the maximum-exposed individual of 10 rem. The estimated frequency for this accident is about once every 50,000 years.

Eight accidents for rail transport of spent fuel were analyzed: three minor, three moderate and two severe. Two minor accidents involved derailments and 30-minute fires; no release occurred. The third minor accident involved undetected leakage of cask coolant. This accident could occur up to twice per year and result in a 70-yr accumulated dose of 2×10^{-5} rem to the maximum-exposed individual.

The moderate accidents involved cask impacts, fire-induced cask venting, and failures in the mechanical cooling system as a result of accident forces. The cooling system failure is estimated to occur once every 50 years and results in a 70-yr accumulated dose of 8×10^{-5} rem to the maximum-exposed individual.

Severe accidents resulting from extreme impacts and a prolonged loss of cooling to a design load of fuel assemblies could release significant amounts of radioactive material. Such an accident was estimated to occur once every 50,000 years. Seventy-year accumulated doses to the maximum-exposed individual of 130 rem and 140 man-rem to local populations excluding the maximum-exposed individual would result from such an accident involving 6-month cooled fuel. However, with fuel that has been cooled for several years before shipment (as planned for the once-through fuel cycle), an accident of this severity is not plausible. In a separate study of fuel transportation accidents (DOE/EIS-0015), it is reported that a maximum-exposed individual would receive a 50-yr accumulated dose of only about 0.4 rem from such an accident involving 4-yr cooled fuel (0.6 rem for a 70-yr dose).

4.8.1.2 Radiological Impacts from Unpackaged Spent Fuel Storage Accidents

The example concept for interim spent fuel storage is a 3000-MTHM capacity away-from-reactor storage facility (AFR). Eighteen accidents were postulated for the receipt and storage of unpackaged spent fuel at an AFR: eight minor, seven moderate and three severe. Accident details are described in DOE/ET-0028, Section 5.7. Eight accidents were determined to have potential for release of radioactive material. Four of the eighteen accidents relate to the operation of off-gas systems at the AFR. These accidents are not discussed here because releases from this system would be smaller than accidental releases from the dissolver off-gas system in the fuel reprocessing plant (Section 4.8.2.1) that were designated as the umbrella source terms. (Those releases result in an estimated 70-yr accumulated dose to the maximum-exposed individual of 2×10^{-3} rem.)

Releases resulting from minor accidents were added to expected annual operational releases for this facility based on their estimated frequencies.

Moderate accidents include fuel-handling mistakes, dropped transport casks and uncontrolled venting of rail casks. Releases from these accidents are smaller than those from a packaging facility accident, which is designated as the umbrella source term discussed in Section 4.8.1.3. (Those releases result in less than 3×10^{-5} rem accumulated dose to the maximum-exposed individual during the 70 years after the accident.)

A strike by a design-basis tornado, a criticality event in storage, and a loss of cooling were considered severe accidents at an AFR. The postulated criticality is estimated to occur only once every 100,000 years and results in an estimated 70-yr dose to the maximum-exposed individual of 5×10^{-2} rem.

4.8.1.3 Radiological Impacts Due to Accidents at a Fuel Packaging Facility

A fuel packaging facility (FPF) will be required to prepare fuel for disposal in the once-through cycle. The fuel packaging facility may be colocated with either the AFR, a packaged fuel storage facility or a spent fuel disposal facility. Radiological impacts that result from accidents at the packaging facility are not dependent on its location.

Six accidents were postulated for spent fuel packaging operations: three minor, two moderate and one severe. The three minor accidents involve minor fuel-handling equipment failures and are expected to result in no releases of radioactive material.

A dropped fuel element occurring about once per year was considered a moderate accident. The 70-yr dose to the maximum-exposed individual from this accident was estimated to be less than 1×10^{-5} rem.

A worst-case fuel drop accident, in which the cladding on 20% of the fuel rods is ruptured, was estimated to occur once every 100 years. This severe accident is estimated to result in less than 3×10^{-5} rem accumulated dose to the maximum-exposed individual during the 70 years after the accident.

4.8.1.4 Radiological Impacts from Packaged Spent Fuel Storage Accidents

If spent fuel is to be stored for extended periods before disposal, it may be desirable to store it as packaged spent fuel. Accidents at such facilities are discussed here. Accidents for the handling of spent fuel at a waste repository are discussed in Section 5.5.

Representative accidents for packaged spent-fuel receiving operations were considered to be similar to those postulated for a spent-fuel packaging facility (Section 4.8.1.3).

Four technologies were considered for the extended storage of packaged spent fuel: one wet and three dry. A water basin concept was considered for wet storage. Dry storage was considered in vaults, caissons and surface casks.

Nine accidents were postulated for the water basin storage of packaged fuel. Six are the result of the loss of essential basin services and would cause no release. A strike by a design-basis tornado or a criticality in the pool were considered to be severe accidents, but are expected to release less radioactivity to the environment than the equivalent accidents in the pool storage of unpackaged fuel discussed in Section 4.8.1.2 (a 70-yr dose to the maximum-exposed individual of 5×10^{-2} rem).

Various sets of severe environmental conditions were postulated for the dry storage concepts. No design-basis environments were considered capable of causing a release of radioactive material. Package failures resulting from unidentified defects or corrosion were the only mechanisms identified for material releases from dry storage. Releases are estimated to occur once every 10 years from the example facility and result in a 70-yr accumulated dose to the maximum-exposed individual of 1.1×10^{-6} rem.

4.8.1.5 Non-Radiological Impacts of Accidents in the Once-Through Cycle

Disabling injuries and deaths will result from construction of waste management facilities, as they do in construction of all facilities. Using estimates of man-hours involved in facility construction and statistical injury and death rates for construction activities (13.6 disabling injuries and 0.17 deaths per million man-hours), we estimate that 110 disabling injuries and less than two deaths will result from construction of a 3000 MTHM AFR

with a colocated spent fuel packaging facility. About 60% of these injuries and deaths are attributable to the AFR itself, and 40% are attributable to the packaging facility. Decommissioning activities are estimated to result in only about 3% as many deaths and injuries as do the construction activities.

Injuries and deaths will also result from spent fuel transportation, as they do from other transportation activities. For rail transport, we use estimates of 0.36 disabling injuries and 0.039 deaths per million km. For truck transport, the estimates are 0.44 disabling injuries and 0.045 deaths per million km. These injuries and deaths may occur either to the transportation worker or to the public.

4.8.2 Accident Impacts for the Reprocessing Fuel Cycle

This section describes the impacts of postulated accidents in the predisposal waste management operations required in the reprocessing fuel cycle.

4.8.2.1 Radiological Impacts from Accidents During the Treatment and Packaging of Reprocessing Wastes

In the reprocessing fuel cycle, both high-level and TRU wastes are generated at the fuel reprocessing plants (FRP), but only TRU wastes are generated at the fuel fabrication plants (MOX-FFP). Discussions of waste management accidents at these facilities are divided into high-level, transuranic, and gaseous or airborne waste management operations.

Calcination and vitrification processes were considered for the treatment of high-level liquid wastes. Minor and moderate accidents involving in-cell material spills, process equipment failures and the loss of components in the off-gas treatment processes were considered. No credible scenarios for severe accidents were identified for either of these technologies. Accidental releases are, in part, mitigated by processing through the FRP atmospheric protection system (a final exhaust-air filtration system).

The largest release from a minor accident results from a 2-kg calcine spill to the cell. Spills of this magnitude are estimated to occur once in 10 to 1000 years, but smaller spills to the cell probably will occur more frequently. The 70-yr accumulated dose to a maximum-exposed individual from this accident is 6×10^{-6} rem.

A moderate accident involving the loss of an off-gas filter is estimated to occur once every 5 years. The 70-yr accumulated dose to a maximum-exposed individual would be 2×10^{-4} rem for this accident. All other moderate accidents for the high-level waste treatment facilities would result in smaller doses.

Transuranic wastes generated in the example FRP consist of fuel hulls and hardware, failed equipment, combustible and noncombustible wastes and wet wastes. Similar wastes, with the exception of hulls and hardware, are also produced at the MOX-FFP.

Packaging without compaction, hulls compaction and hulls melting were considered for the treatment of fuel hulls and hardware. No credible moderate or severe accidents were identified for any of these technologies. The worst minor accident postulated was a

zirconium fire. In this accident, 2 kg of irradiated zirconium are available for combustion. The 70-yr accumulated dose to a maximum-exposed individual was estimated to be 1×10^{-9} rem.

Failed equipment will be disassembled at both the MOX-FFP and the FRP. It is anticipated that during this operation equipment could tip over or be dropped by an overhead crane. The primary hazard from these accidents is to plant workers. No offsite releases will occur.

Combustible waste treatment technologies involve either packaging with no treatment, or controlled air incineration followed by ash immobilization. Generally, the minimum treatment processes did not have potential for other than minor accidents. Both minor and moderate accidents were identified for controlled air incinerators. No credible severe accidents were identified for the treatment of combustible wastes.

Minor accidents involving combustible wastes include minor ruptures in waste bags, small fires and waste package spills. The consequences of the largest release from a minor accident are a 70-yr accumulated dose to a maximum-exposed individual of 2×10^{-4} rem.

Moderate accidents in the incineration operation include explosions and large fires. The largest 70-yr accumulated dose from a moderate accident is 8×10^{-5} rem to the maximum-exposed individual.

Eight accidents were identified for the immobilization of wet wastes using the bitumen process: six minor and two moderate. Similar accidents are also plausible for the cementation process.

Minor accidents that do not generate aerosols were considered to have no release of material beyond the processing cell area. Spillage of the treated waste product would be contained in the cell. A bitumen fire will result in the largest minor accident release. The impact of releases from this accident would be negligible.

The accident with the largest release, classified as a moderate accident, was a filter failure concurrent with a bitumen fire. This accident is expected to occur about once every 300 years and result in a 70-yr accumulated dose to the maximum-exposed individual of 5×10^{-7} rem.

There are two types of radioactive components in gaseous effluent streams. The first is radioactive gases and volatilized radionuclides. These components are captured either by adsorption beds or by cryogenic processing of the gas stream. The second is radioactive particulates entrained in the gas flow. These particulates are captured by the use of highly efficient filtration systems. Gas effluent air processing systems at the FRP may use all of these processes. However, at the MOX-FFP, filtration is the only process employed since particulates are the only significant materials in the off-gas effluent.

Minor and moderate accidents were identified for the treatment of gaseous waste streams. No credible severe accidents could be identified. Minor accidents include plugged beds and filters, minor leakage through processing equipment and failure of active system components

such as blowers, pumps, etc. These accidents are considered to have no releases sufficient for consideration as an accidental release. Minor leakage was added to normal operating releases.

Moderate accidents include catastrophic filter ruptures, rupture of catalytic units during changeout and shutdown of all treatment systems. The largest release of this type would result from a shutdown of the dissolver off-gas system at the FRP for 30 days. Iodine, ruthenium, carbon and krypton would be released. A maximum-exposed individual is estimated to receive a 70-yr accumulated dose of 3×10^{-2} rem from this accident. The accident is estimated to occur about once every 10 years.

4.8.2.2 Radiological Impacts from Reprocessing Waste Storage Accidents

If waste disposal facilities are not available at the time wastes are being generated, interim storage will be required. Several storage alternatives have been analyzed for high-level waste, TRU waste, and krypton.

At the example FRP, high-level waste is solidified immediately after generation. Canisters of solidified high-level waste are then stored in water basins until they have aged sufficiently for disposal (5 years assumed). If a disposal facility is not available at that time, the waste is assumed to be sent to a sealed-cask interim surface storage facility.

Fifteen accidents were identified for water basin storage of solid high-level waste: six minor, five moderate and four severe.

Minor accidents include failure of components in ventilation and cooling systems. No releases result from these accidents.

Moderate accidents include failures of basin structural components, canister handling errors and canister failure during storage. No releases to the environment result from these accidents. Increased worker exposures are expected for accidents that release activity to the pool water.

Severe accidents in this facility involve dropping large objects into the pool, fires and a design-basis tornado strike. Consequences of these accidents are less than those cited in Section 4.8.1.2 for a spent-fuel storage pool (a 70-yr dose to the maximum-exposed individual of 5×10^{-2} rem).

The only accident with a potential for environmental consequences during sealed-cask storage of solidified high-level waste is a canister rupture during its placement in a storage cask. The accident is considered of moderate severity and, using calcine, would result in a 70-yr accumulated dose to the maximum-exposed individual of 8×10^{-3} rem.

Transuranic wastes include drums and boxes of contact-handled TRU wastes and drums and canisters of remotely handled TRU wastes, including packaged fuel residues. No credible severe accident scenarios were identified for TRU waste storage. Accidents for the storage of fuel residue are all less severe than accidents described for the cask storage of solidified high-level waste. Outdoor storage methods for all TRU wastes and indoor storage

methods for remotely handled TRU wastes have potential for both minor and moderate accidents. Indoor storage methods for contact-handled TRU wastes limit the accident spectrum to minor accidents.

Typical minor accidents involving TRU waste packages include dislodging of surface contamination, rusting through of containers, and mechanical breaching of package. The 70-yr accumulated dose for the maximum-exposed individual for the largest of these releases is 2×10^{-4} rem.

Moderate accidents include fires in storage, tornado strikes and drums dropped from a crane. The 70-yr accumulated dose to the maximum-exposed individual for the largest of these releases is 4×10^{-4} rem.

Krypton removed from the FRP dissolver off gas is assumed to be collected in pressurized gas cylinders and stored onsite at the FRP in a separate facility. Three moderate accidents were postulated for the release of gas from one cylinder (130 kCi). If this occurs in the operating area or storage corridor, gas would be released via the facility stack. The 70-yr accumulated dose to a maximum-exposed individual in the public would be 5×10^{-3} rem. This accident is estimated to occur once every 20 years. Of greater potential consequence are the employee doses from this accident. A worker in the area of the ruptured cylinder faces hazards from flying debris and could receive a radiation dose rate of up to 8 rem/min. Immediate evacuation of the area would be required.

4.8.2.3 Radiological Impacts from Reprocessing Waste Transportation Accidents

A reprocessing fuel cycle has potential transportation requirements for spent fuel, solidified high-level waste, fuel residues, and other TRU wastes. As in the once-through cycle, safety during transport depends primarily on shipping containers. The containers must meet standards established by the Department of Transportation and the Nuclear Regulatory Commission. Packages containing significant amounts of radioactive material must be designed to prevent loss or dispersal of the radioactive contents, retain shielding efficiency, ensure nuclear criticality safety, and provide adequate heat dissipation under normal conditions of transport and under specified (hypothetical) accident damage test conditions (49 CFR 71, Appendix B). Improbable accidents that exceed the hypothetical tests, accidents due to equipment failures and accidents that are less severe than the test conditions were considered here to demonstrate the range of potential occurrences in a transportation environment.

Minor, moderate and severe accidents were postulated for the rail transport of solidified high-level waste. Minor accidents for this material are similar to those for spent fuel. A moderate accident could result in a reduction in neutron shielding and a local hazard of increased neutron exposures. No radioactive material would be released in this accident. A severe accident involving impact and fire could result in a material release. This accident is estimated to occur only once every 330,000 years and result in a 70-yr accumulated dose to the maximum-exposed individual of 10 rem.

Transuranic wastes were considered to be transported in DOT-licensed packages. Three minor and one severe accident were identified. The worst minor accident is expected to

occur once per year due to improperly closed waste packages and result in a 70-yr accumulated dose to the maximum-exposed individual of 3×10^{-3} rem. A severe accident involving severe impact and fire with an estimated frequency of once every 100,000 years would result in a maximum-exposed individual 70-yr whole body dose of 3 rem.

4.8.2.4 Non-Radiological Impacts of Accidents in the Reprocessing Cycle

Estimates of deaths and disabling injuries resulting from construction and decommissioning of reprocessing fuel cycle waste management facilities are given in Table 4.8.1. Injuries and deaths also result from transportation of the wastes. As in spent fuel transport, we use estimates of 0.36 disabling injuries and 0.039 deaths per million km for rail transport and 0.44 disabling injuries and 0.045 deaths per million km for truck transport. These injuries and deaths may occur either to the transportation worker or to the public.

TABLE 4.8.1. Disabling Injuries and Deaths from Construction and Decommissioning of Reprocessing Fuel Cycle Waste Management Facilities

Construction	(a) Disabling Injuries	(b) Deaths
Waste Mgmt. Facilities at Example FRP	55	0.7
Waste Mgmt. Facilities at Example MOX-FFP	5	0.06
Example RWSF	415	5
Decommissioning		
Waste Mgmt. Facilities at Example FRP	25	0.3
Waste Mgmt. Facilities at Example MOX-FFP	5	0.06

(a) Based on frequency rate of 13.6 per million man-hours.

(b) Based on frequency rate of 0.17 per million man-hours.

4.8.3 Radiological Impact Summary for Predisposal Operations Accidents

Table 4.8.2 summarizes the radiation effects of the predisposal-system accident analysed for this Statement.

This comparison shows that transportation is the waste management step with the potential for the most serious accident in either fuel cycle. The estimated exposures in these accidents, however, are not large enough to cause observable clinical effects. The individuals exposed would presumably bear an increased probability of developing cancer sometime during their life or of passing on a genetic defect.

TABLE 4.8.2. Summary of Radiation Effects from Potential Worst-Case Predisposal System Accidents

	<u>Maximum-Exposed Individual Radiation Doses, rem</u>	
	<u>Once-Through Cycle</u>	<u>Reprocessing Cycle</u>
Transportation		
Spent Fuel (4-yr-old)	0.6(a)	
HLW		10(b)
TRU Waste		3
Storage	5×10^{-2}	8×10^{-3}
Treatment and Packaging	3×10^{-5}	2×10^{-3}

(a) Shipment of 6-month-old spent fuel, which is unlikely, could result in a maximum dose of 130 rem.

(b) Based on HLW 6.5 years after reactor discharge.

REFERENCES FOR SECTION 4.8

Code of Federal Regulations, 49 CFR-173.398.

U.S. Department of Energy. 1979. Technology for Commercial Radioactive Waste Management, DOE/ET-0028, Washington, D.C.

U.S. Department of Energy. 1979. Environmental Aspects of Commercial Radioactive Waste Management, DOE/ET-0029, Washington, D.C.

U.S. Department of Energy. 1979. Final EIS on U.S. Spent Fuel Policy for the Storage of U.S. Spent Power Reactor Fuel, DOE/EIS-0015, Washington, D.C.

4.9 COST OF PREDISPOSAL OPERATIONS

Costs for treating, storing, and transporting spent fuel or commercial reprocessing and mixed oxide fuel fabrication wastes are presented in this section. All costs are stated in terms of constant^(a) 1978 dollars.

The costs shown here are levelized^(b) unit costs based on capital, operating, and decommissioning costs for the individual predisposal waste management operations. Capital, operating, and decommissioning cost estimates have been developed as part of this Statement for the predisposal facilities associated with the example geologic disposal system and are summarized in Appendix A. Predisposal costs for alternatives other than geologic disposal are based on predisposal costs of the geologic disposal system where the operations are similar. Where the operations are different, data from other studies have been used to the extent available.

For the once-through cycle, the mined geologic and very deep hole concepts have the lowest predisposal systems costs (\$103/kg HM) of the alternatives studied in this Statement. Costs of other alternatives are 50 to 100% higher. For the reprocessing cycle, the mined geologic, very deep hole, well injection, space injection, and rock melting alternatives all cost about \$170/kg (including spent fuel storage and transportation). Costs of other alternatives ranged from \$15 to over \$230/kg HM more than the lowest cost options.

The cost tables in this section are intended to provide predisposal cost comparisons between disposal alternatives and to illustrate cost relationships among predisposal components for the example geologic disposal alternative. The total costs presented here do not include the significant costs of research and development. Costs for the entire waste management system, levelized with respect to the power generation that produced the waste, are developed in Chapter 7.

A brief explanation of the cost estimate assumptions and bases for the costs developed in this Statement is given in Section 3.2. Additional detail on predisposal facility costs for geologic disposal is available in DOE/ET-0028, Volumes 2, 3 and 4.

4.9.1 Once-Through Fuel Cycle Predisposal Costs

For the example once-through cycle, predisposal operations consist of storage at reactor basins, storage in independent basins when reactor basin capacities are exceeded, treatment and packaging of the fuel assemblies, and all transportation operations. A brief description of the operations required for each disposal option is found in Table 4.1.1.

Table 4.9.1 lists the costs associated with these predisposal operations for the alternative disposal methods studied. Reactor basin storage charges of \$25/kg HM and transportation costs of \$26/kg HM for shipment of spent fuel to treatment facilities are common to all

(a) For a definition of constant dollar costs, see Section 3.2.8.1.

(b) Levelizing refers to developing a single, constant unit charge, which recovers all expenditures associated with a facility or system including interest (see Section 3.2.8.2). The costs stated in this section are levelized with respect to individual waste management operations only.

TABLE 4.9.1. Unit Costs of Predisposal Operations for Once-Through Cycle Disposal Options

Predisposal Operation	Cost, \$/kg HM								
	Mined Geologic	Very Deep Holes	Rock Melting	Island	Sub-seabed	Ice Sheet	Injection Well	Transmutation	Space Injection
5-Year Reactor Storage	25	25	25	25	25	25	25	25	25
Shipment to Interim Storage (1000 mi) ^(a)	5	5	9 ^(c)	5	5	5	9 ^(c)	9 ^(c)	9 ^(c)
Interim Storage ^(a)	29	29	39 ^(c)	29	29	29	39 ^(c)	39 ^(c)	39 ^(c)
Shipment to Treatment (1500 mi)	26	26	26	26	26	26	26	26	26
Treatment and Packaging	18	18	70 ^(c)	18	18	18	70 ^(c)	200 ^(c)	~100 ^(c)
Shipment to Disposal	-- ^(b)	-- ^(b)	6	50	50	50	6	20	<15
TOTAL	103	103	175	150	150	150	175	320	<214

(a) Based on interim storage of 25% of total spent fuel discharges.

(b) No cost is shown for this step since the analysis assumes that packaging or treatment is accomplished at the disposal site. If packaging facilities for mined geologic disposal of spent fuel were located offsite, an additional transportation step would be necessary for this option.

(c) Includes costs of managing TRU wastes generated during dissolution of the spent fuel.

disposal alternatives. The rock melting, well injection, transmutation, and space injection alternatives have somewhat higher costs for shipment to interim storage, interim storage, and treatment since the spent fuel is dissolved and management of additional waste streams is required. The high transportation costs of the island, subseabed, and ice sheet alternatives are a result of the required land and ocean transportation.

The mined geologic and very deep hole concepts have significantly lower predisposal costs than the other alternatives, \$103/kg HM. The island, subseabed, and ice sheet alternatives have higher costs, \$150/kg HM, because of the expensive transportation requirements. The other alternatives have higher predisposal costs because of the cost of managing the additional waste streams generated. These range from \$175/kg HM for the rock melting and well injection alternatives to \$320/kg HM for transmutation.

4.9.2 Reprocessing Fuel Cycle Predisposal Costs

A brief description of the predisposal operations for the reprocessing fuel cycle required for each of the disposal options is found in Table 4.1.2. Costs associated with these operations are shown in Table 4.9.2. Spent fuel storage and transportation costs could be considered as reprocessing costs rather than as waste management costs if spent fuel is reprocessed. For consistency and conservatism, the costs of spent-fuel storage and shipment are included as waste management costs in this Statement. Without these costs, the predisposal costs of the reprocessing cycle alternatives are comparable to or less than the once-through cycle costs.

Waste treatment costs of the reprocessing cycle alternatives are comparable with two exceptions: 1) costs for the rock melting and well injection alternatives are lower since high-level waste solidification is not required, and 2) costs for the transmutation alternative are higher because of repeated chemical partitioning and target fabrication operations.

Transportation costs for the rock melting and well injection alternatives are less than other options since the high-level waste is not transported offsite. However, the cost of interim storage of the high-level liquid waste for these two alternatives is much higher than the cost of solidified high-level waste storage employed in the other alternatives. Transportation costs for the island, subseabed, and ice sheet alternatives are significantly higher than for other alternatives because of the oceanic shipments of high-level waste.

Total predisposal system costs of the mined geologic, very deep hole, rock melting, well injection, and space injection alternatives are similar, e.g., \$168/kg HM. The costs of the island, subseabed, and ice sheet alternatives are 185/kg HM or about 10% higher and costs of the transmutation alternative (>\$400/kg HM) are more than 100% higher than any other alternative.

TABLE 4.9.2. Unit Costs of Predisposal Operations for Reprocessing Waste Disposal Operations

Predisposal Operation	Cost, \$/kg HM								
	Mined Geologic	Very Deep Holes	Rock Melting	Island	Sub-seabed	Ice Sheet	Injection Well	Transmutation	Space Injection
Spent Fuel Storage and Shipment	59	59	59	59	59	59	59	59	59
Waste Treatment									
• FRP ^(a,c)	67	67	43	67	67	67	43	>230 ^(f)	~67 ^(e)
• MOX-FFP ^(b)	4	4	4	4	4	4	4	>70 ^(f)	4
Shipment to Interim Storage (1000 mi)	6	6	4	6	6	6	4	6	6
Interim Storage ^(d)	23	23	52	23	23	23	52	23	23 ^(e)
Shipment to Disposal (1500 mi)	9	9	6	26	26	26	6	>12	<15
	<u>168</u>	<u>168</u>	<u>168</u>	<u>185</u>	<u>185</u>	<u>185</u>	<u>168</u>	<u>>400</u>	<u><174</u>

(a) Fuels Reprocessing Plant. See Appendix A for a breakdown of example FRP waste treatment costs and options.

(b) Mixed Oxide Fuel Fabrication Plant. See Appendix A for a breakdown of example MOX-FFP waste treatment costs and options.

(c) Includes HLW and TRU waste treatment costs (\$/kg HM) as follows:

	Mined Geologic and Similar Cost Options	Rock Melting	Injection Well	Transmutation	Space Injection
HLW	24	--	--		~24
TRU Waste	43	43	43		43
TOTAL	67	43	43	>>230	~67

(d) A \$10/kg HM cost for TRU waste storage is common to all options. The remaining cost is for HLW storage.

(e) HLW storage costs for those options may differ from those for the mined geologic option because of different configurations. No difference is assumed here.

(f) Based on additional partitioning facility costs.

4.9.3 Detailed Predisposal Cost Estimates for Geologic Disposal

This section describes in greater detail the predisposal cost estimates for the example geologic disposal alternative. Costs are derived for both the once-through and reprocessing cases.

4.9.3.1 Once-Through Fuel Cycle

Table 4.9.3 lists the costs associated with once-through predisposal operations. Reactor basin storage is estimated to cost about \$6/kg HM per year with storage periods on the order of five years, for an equivalent present-worth cost of about \$25/kg HM.

After storage, the fuel assemblies may be: 1) packaged intact, 2) disassembled and packaged, 3) chopped, voloxidized, and packaged, or 4) chopped, the fuel dissolved, and converted to glass. Treatment costs shown in Table 4.9.3 for the above options range from \$18 to \$92/kg HM due to the increasing complexity of these operations.

Costs for independent unpackaged water basin storage of spent fuel vary significantly with the size and capacity utilization of the facility. Costs for storage in a non-expandable 3000 MTHM basin are estimated at about \$117/kg HM.^(a) Costs for a 5000 MTHM non-expandable basin (DOE 1978), using unit cost assumptions in this Statement are estimated at \$80/kg HM.^(a) Estimates for a facility expandable to 20,000 MTHM are \$45/kg HM.^(a) In addition, costs vary nearly inversely with capacity utilization. For example, if a facility utilized only 50% of its capacity, unit costs would be almost doubled.

Other storage options include storage of packaged spent fuel. In these cases, spent fuel could be packaged in facilities located adjacent to storage facilities. Table 4.9.3 illustrates costs for four such design concepts. Dry well storage appears to be the most cost effective alternative.

Packaging of the spent fuel could be done either at facilities adjacent to storage basins or at the repository. Packaging facilities that are integral with the repository are assumed for the example system here and may be more cost effective due to lower transportation costs for unpackaged spent fuel.

Transportation costs include transport of the spent fuel from reactor storage to independent storage (25% of the fuel), reactor storage to repository (75% of the fuel) and independent storage to repository (25% of the fuel).

Total predisposal costs for the example case in Table 4.9.3 are about \$103/kg HM. The range is estimated using the lowest and highest cost options.

(a) In the cases shown in Table 4.9.3, it is assumed that only about 25% of total spent fuel discharges are sent to independent storage and the cost is reduced proportionally.

TABLE 4.9.3. Predisposal Unit Costs for the Once-Through Cycle

Treatment	Unit Cost, \$/kg HM(a)	
	Example System	Other Options
Decay Storage at Reactor Basin	25 ^(b)	--
Package Intact Fuel Assemblies	18	--
Disassemble and Package Fuel Rods	--	38 ^(c)
Package Chopped and Voloxidized Fuel	--	42 ^(c)
Dissolve Fuel and Convert To Glass	--	92 ^(c)
Independent Away-from Reactor (AFR) Fuel Storage		
Unpackaged		
● Nonexpandable 3000 MT Basin	29 ^(d)	--
● Nonexpandable 5000 MT Basin	--	20 ^(d)
● Modular Basin Expanded to 20,000 MT	--	11 ^(d)
Packaged		
● Water Basin	--	38
● Air-Cooled Vault	--	35
● Dry Well	--	22
● Surface Cask	--	30
Transportation	31 ^(d,e)	--
Total	103	range 85 to 186

- (a) Costs may be expressed in \$/GWe-yr by multiplying by 38,000 kg HM/GWe-yr.
- (b) Reactor basin spent fuel storage costs are based on a charge of \$6/kg HM per year. The value shown in the table is equivalent to a minimum storage time of 5 years with a real cost of money of 7% per year.
- (c) Estimates based on facilities and operations described in ONWI-39, July 1979, except that the cost calculations were modified to a 7% real cost of money basis. Estimates include treatment of all wastes generated, but do not include transportation and disposal. Costs for the entire system are shown in Table 4.9.7.
- (d) Average fuel cycle cost based on interim storage of 25% of total spent fuel discharges.
- (e) Packaging may be done at the repository or at another site. The transportation costs for the example case are based on a packaging facility which is integral with the repository and assumes that packaged fuel handling is accomplished using repository facilities. Transportation costs consist of \$5/kg HM for shipment of 25% of the spent fuel to AFR storage, \$20/kg HM for shipment of the other 75% of the spent fuel from reactor basins to final disposal and \$6/kg HM for shipment of the fuel in AFR storage to final disposal.

4.9.3.2 Reprocessing Fuel Cycle

Reprocessing fuel cycle wastes consist of wastes from reprocessing and mixed oxide fuel fabrication plants. Table 4.9.4 shows the unit costs for alternative methods of waste treatment for these wastes.

Differences in cost between treatment options are not large, ranging from 10 to 25%, except for krypton removal. Predisposal costs for the example system are fairly evenly distributed between high-level waste treatment (\$23.9/kg HM), TRU waste treatment (\$18.40/kg HM), gaseous waste treatment (\$28.20/kg HM), interim storage of high-level and TRU wastes (\$23.10/kg HM) and transportation (\$15.50/kg HM). These costs total about

TABLE 4.9.4. Unit Cost Estimates for Reprocessing Fuel Cycle Wastes

Waste Category	Treatment	Unit Cost, \$/kg HM ^(a)	
		Example System	Other Options
High-Level Liquid Waste	Spray Calcination & Vitrification	10.4	
	Fluid Bed Calcination Only		13.0
	5-Year Onsite Storage & Handling (after solidification)	13.5	
Fuel Residue	Package Without Compaction (in sand)	4.9	
	Compaction of Hulls		4.6
	Melting of Hulls		5.1
Non-Combustibles and Failed Equipment	Package	4.8(b)	
Combustible and compactable	Incineration	4.4(b)	
	Package Only		3.7(b)
Wet Waste	Cementation	4.3(b)	
	Bitumenization		4.3(b)
Gaseous Waste	Vessel Off-Gas	3.9	
	Dissolver Off-Gas		
	I and Ru Removal		2.0
	I, Ru and C Removal		3.2
	I, Ru and Kr Removal		6.0
	I, Ru and C and Kr Removal	6.1	
	Kr Storage Onsite	16.4	
	Atmospheric Protection System (APS)		
	Group III Prefilter	1.8	
Sand Filter		3.8	
Deep Bed Filter		2.5	
Solidified Reprocessing Wastes	Interim Storage ^(c)	23.1	
	Transportation ^(d)	<u>15.5</u>	
	Subtotal	109	
	Spent Fuel Storage and Shipment Prior to Reprocessing	<u>59(e)</u>	
Total		168	range 139 to 179

(a) Costs may be expressed in \$GWe-yr by multiplying by 38,000 kg HM/GWe-yr.

(b) Includes estimates for waste treatment at the mixed oxide fuels fabrication plant. See Appendix A for further detail.

(c) See Table 4.9.5.

(d) See Table 4.9.6.

(e) Based on a 1-year storage of all spent fuel at the reactor basin (\$6/kg HM) and interim storage of 25% of total spent fuel discharges (\$29/kg HM). Spent fuel transportation is estimated to cost \$24/kg HM (see Table 4.9.6). Although spent fuel handling and storage prior to reprocessing are not clearly waste management functions, the costs are shown here and are included in the systems cost estimates in Chapter 7 to conservatively estimate waste management costs.

\$109/kg HM, which is comparable to the \$103/kg HM predisposal cost totals for spent fuel waste management. In addition, spent fuel handling and storage costs before reprocessing are also included for reasons noted previously, bringing the total reprocessing fuel cycle waste management cost to \$168/kg HM. The range is estimated using the lowest and highest cost treatment options.

Tables 4.9.5 and 4.9.6 show additional detail for the costs of interim storage options and transportation operations.

4.9.4 Detailed Subsystem Costs for Geologic Disposal

Since many treatment options affect the treated waste volumes, the entire cost impact of these options cannot be evaluated on the basis of the predisposal costs alone. For this reason final disposal costs are included in the subsystem costs presented here, although they are not developed in this Statement until Section 5.6.

Table 4.9.7 illustrates total subsystem waste management costs for waste management operations for both the once-through and reprocessing fuel cycles. These costs include the effect of volume reduction on subsequent transportation, interim storage and disposal operations. For the once-through cycle, dissolving the spent fuel costs significantly more than other treatment options. For the reprocessing cycle, treatment options do not have a significant impact on total system costs except for the fuel residue and combustible waste options. The high cost of removing and storing krypton relative to the waste management costs of removing other gases can also be noted.

The cost ranges reflect the impact of volume changes on costs assuming the example interim storage and final disposal methods. The upper cost estimate assumes the least volume reduction and the lowest cost estimate the greatest volume reduction. Cost ranges would be somewhat greater than shown here for other interim storage and final disposal options.

The example total cost estimate of \$215/kg HM for the reprocessing fuel cycle includes \$59/kg HM for spent fuel transportation and storage prior to reprocessing. The final disposal costs included in the subsystems cost estimates may be estimated by subtracting the predisposal costs in Tables 4.9.3 and 4.9.6 from the subsystem cost in Table 4.9.7 for the once-through and reprocessing fuel cycles.

TABLE 4.9.5. Unit Cost Estimates for Interim Storage Operations for Reprocessing Fuel Cycle Wastes

Waste Category	Operation	Unit Cost \$/kg HM ^(a)	
		Example	Option
High-Level Waste	Sealed Cask Storage	13	--
Fuels Residue and Other TRU Waste Canisters	Dry Well Storage Vault Storage	7	20
Remotely Handled TRU Waste Drums	Vault Storage Dry Well Storage	3	6
Contact-Handled TRU Waste Drums and Boxes	Outdoor Surface Storage Indoor Unshielded Storage	0.3	0.4
Total		23	

(a) Costs may be expressed in \$/GWe-yr by multiplying by 38,000 kg HM/GWe-yr.

TABLE 4.9.6. Unit Cost Estimates for Example Transportation Operations, \$/kg HM^(a)

Origin and Destination	Unit Cost for Spent Fuel	Unit Cost For Reprocessing Wastes			
		High-Level Waste	Fuel Residue Waste	Other RH-TRU Waste	CH-TRU Waste
Reactor to Interim Storage (1000 mi)	5(b)	--	--	--	--
Reactor to Reprocessing Plant (1000 mi)	14(c)	--	--	--	--
Interim Storage to Reprocessing Plant (1000 mi)	5(b)	--	--	--	--
Reprocessing Plant to Interim Storage or Repository (1000 mi)	--	2.0	2.2	1.7	0.2
Interim Storage to Repository (1500 mi)	--	<u>3.0</u>	<u>3.5</u>	<u>2.6</u>	<u>0.3</u>
Total	24	5	6	4	0.5

(a) Costs may be expressed in \$/GWe-yr by multiplying by 38,000 kg HM/GWe-yr.

(b) Based on interim storage of 25% of the spent fuel.

(c) Based on direct shipment of 75% of the spent fuel.

TABLE 4.9.7 Subsystems^(a) Waste Management Costs for Alternative Waste Treatment Options

Fuel Cycle Option	Waste Category	Option	Systems Cost, ^(a) \$/kg HM ^(b)			
			Example	Options		
Once-Through	Spent Fuel	Encapsulate Whole Assemblies	155			
		Disassemble and Encapsulate		~140		
		Chop Assemblies and Encapsulate		~150		
		Dissolve Fuel, Convert to Glass and Encapsulate		~250		
	Total		155 ^(c)	range 140 to 250		
Reprocessing	High-Level	Vitrification	66			
		Calcination		69		
	Fuel Residue	Package in Sand Without Compaction	20			
		Compaction of Hulls		14		
		Melting of Hulls		11		
	Non-combustible and Failed Equipment	Package	20			
	Combustibles	Incinerate Package Only	10	41		
	Wet	Cementation	12			
		Bituminization		8		
	Gaseous	Vessel Off-gas Dissolver Off-gas	<ul style="list-style-type: none"> ● I and Ru Removal ● I, Ru and C Removal ● I, Ru and Kr Removal ● I, Ru, C and Kr Removal 	4		
						2
						3
						22
					22	
	Atmospheric Protection System	2				
	● Group III Prefilter					
	● Sand Filter		4			
	● Deep-Bed Filter		3			
	Subtotal		156			
	Spent Fuel	Storage and Transportation Prior to Reprocessing	59			
	Total		215 ^(d)	range 182 to 251		

(a) Subsystems costs include the cost of waste treatment, packaging, all transportation, interim storage and final disposal. Research and development costs and the discount rate effect of timing of the costs are not included in the figures shown here, but are included in the system power cost estimates in Chapter 7.

(b) Costs may be expressed in \$/GWe-yr by multiplying by 38,000 kg HM/GWe-yr.

(c) Includes \$52/kg HM for geologic disposal.

(d) Includes \$47/kg HM for geologic disposal.

REFERENCES FOR SECTION 4.9

Office of Nuclear Waste Isolation. 1979. An Assessment of LWR Spent Fuel Disposal Options, ONWI-39, Office of Nuclear Waste Isolation, Battelle-Columbus Laboratories, Columbus, Ohio.

U.S. Department of Energy. 1978. Conceptual Design Report for the Away from Reactor Spent Fuel Storage Facility, DPE 3547. Prepared by E.I. Du Pont De Nemours and Co., Inc. Wilmington, Delaware.

4.10 SAFEGUARDS INCLUDING PHYSICAL PROTECTION FOR PREDISPOSAL OPERATIONS

Regulations similar to those already in place to protect the public from theft of nuclear material and from sabotage at licensed nuclear facilities are expected to apply to operations at waste management facilities. The probable safeguard requirements for predisposal waste management facilities are described in this section.

4.10.1 Safeguards Requirements for the Once-Through Cycle

Safeguards measures, including physical protection, required for currently licensed nuclear facilities are expected to be adequate for safeguards and physical protection for the once-through cycle. Spent fuel and the facilities designed to manage this material are not expected to require additional safeguards.

4.10.1.1 Spent Fuel Treatment and Packaging Safeguards Requirements

The susceptibility of the spent fuel handling operation to theft and sabotage of the fuel elements is reduced as packaging and treatment operations of the fuel elements proceed. The spent-fuel elements and all treatment and packaging facilities handling this material will be physically protected as required by Federal regulations for vital areas (see Section 3.2.9 or 10 CFR 70, 73). All of the auxiliary systems for spent fuel handling will be similarly protected because they are part of the same facility.

If the spent fuel is simply encapsulated for disposal as in the example process for this Statement, the spent fuel elements become less attractive and less accessible targets for sabotage. In addition, operating safety features inherent in the design of facilities licensed to process spent fuel elements contribute significantly to safeguarding this material.

If the spent fuel is chopped and encapsulated, none of the additional steps required in this process significantly increase the susceptibility of the facility, equipment or target material to theft or sabotage.

If the spent fuel is dissolved and converted to glass, the physical protection requirements and the relative unattractiveness and inaccessibility of the material make it an unlikely target for theft or sabotage. The same protective environmental and control measures identified above are present in this facility to provide required safeguards features.

4.10.1.2 Safeguards Requirements for Spent Fuel Storage

Spent fuel is neither easily accessible nor an attractive enough source of fissile material to encourage theft. The plutonium concentration is low and the fuel elements are very radioactive; massive shielding of steel, lead, concrete or several feet of water is required at all times. Separation of the plutonium requires complex chemical processes carried out in remotely operated, shielded processing equipment. In addition, spent fuel is not in a form suitable for easily dispersing radioactive material, and thus, is not an attractive target for this threat because only intact spent fuel rods are considered to be an acceptable form for extended storage.

Physical protection features required by Federal regulations are expected to provide adequate safeguards. Safeguards contingency plans in these regulations for licensed facilities will include NRC-approved arrangements for support from local law enforcement personnel if there is a serious threat. An adequate response force will be able to engage and contain the intruders in less time than is required for the intruders to gain access, remove fuel elements from the storage location, transfer them to a shielded container, place them on a vehicle and leave the site. A single fuel assembly weighs more than one-quarter metric ton and a hoist or crane operated from behind heavy shielding is required to move it. Disassembly to obtain individual fuel rods, which could be transferred by more readily obtainable light equipment, would be a much more time-consuming operation. The disassembly would have to be done remotely, behind heavy shielding or under water.

These same measures also deny fuel storage facility access to saboteurs. A detailed study (Voiland et al. 1974) of the safeguards risks associated with water basin storage of spent fuel concluded that the stored irradiated fuel at the facility under consideration is not amenable to a credible sabotage event that would endanger the public health and safety. The safeguards measures assumed for that case are typical of those required for the licensed facilities.

4.10.1.3 Safeguards Requirements for Transport of Spent Fuel

Spent fuel is more vulnerable to theft and acts of sabotage during transport than at fixed sites because it is more accessible. The measures proposed to protect against diversion and sabotage of shipments of spent fuel reflect this potential threat (10 CFR 73).

The level of physical protection required for shipments of spent fuel elements, established by the NRC in an interim rulemaking (10 CFR 73 1979), was based on a study by Sandia laboratories (1977). Specific requirements were included to protect the public against sabotage of spent fuel in transit by truck or rail, with particular concern for urban areas.

Theft of spent fuel to obtain the fissionable material is not sufficiently credible to warrant additional requirements for this specific threat (see Section 4.10.1.2). Theft of this material as a part of an extortion attempt would be limited to the length of time law enforcement personnel would need to locate the stolen cask. Such material in a cask is detectable by aerial radiation surveys and the fact that detection would be imminent would deter any lengthy extortion scheme.

Prediction of detection is based upon the capability of the Department of Energy's Aerial Radiological Monitoring system (ARMS) of which two are in continuous service (Doyle

1976). It is assumed that one of these or an equivalent system would be available. The system consists of a forty-sensor array with a computer-assisted data analyzer, a printer and a plotter mounted in a helicopter.^(a)

4.10.2 Safeguards Requirements for the Reprocessing Fuel Cycle

In the reprocessing fuel cycle large quantities of fissionable and radioactive material are handled in a fuel reprocessing facility, and the physical protection requirements for the facility and vital materials within it are specified in 10 CFR 73. The general features of those requirements are identified in Section 3.7. Similar requirements would be enforced at the plutonium-uranium mixed oxide fuel fabrication plants.

The waste materials produced at these facilities are unattractive as targets of theft compared to the fissionable material in the facilities. In addition, all waste treatment operations and storage of highly radioactive wastes would be protected in "vital" areas. Consequently, these materials would be inaccessible to any but authorized persons, and successful intrusion, theft and sabotage are improbable.

4.10.2.1 Safeguards Requirements for the Treatment of Reprocessing Wastes

High-level waste is not a potential source of fissionable material and could only be a target for theft or sabotage to disperse or threaten dispersal of radioactive material. The HLW is an unattractive target because of its high radiation level and inaccessibility. All handling, storage, and treatment in the facility occurs by remote operations in shielded, isolated vessels and cells.

Before it is solidified, HLW may be stored as a solution in shielded tanks in which it is accessible only by remote means. Its intense radioactivity and high heat release rates and the maze of facility support equipment would make unauthorized transfer of HLW to a shielded container and its removal offsite an incredible accomplishment, particularly since extensive physical protection measures would also have to be overcome. For similar reasons, dispersal of HLW onsite by explosives is not credible, although the concentration of radioactive fission products in this waste may make it appear to be an attractive target.

With inside assistance, physical protection and access control measures could possibly be compromised, and sabotage of the storage facility could occur. One consequence could be a disruption of the waste cooling system and/or electrical system. Self-heating would cause the contents to begin to boil in about 7 hours and boil to dryness in about 100 hours. This scenario is not considered credible if the planned safeguards measures and the safety design features of the facility (which are included to ensure continuity of HLW cooling) are

(a) The ability of an aerial radiation survey to detect a spent fuel cask that has not been breached and is located inside a facility depends upon the facility. In a single-storied, conventionally constructed warehouse or its structural equivalent, the radiation from the cask would be readily detected in an aerial survey. If the truck and fuel cask were in an underground garage under a multi-storied building surrounded by multi-storied buildings, an aerial survey may not be effective. However, a mobile surface survey would be effective in detecting this source.

considered. Some facility damage and a 300-l (80-gal) spill to the ground during a 3-hour period are considered to be representative of the most serious results from the worst act of sabotage (see DOE/ET-0028, p. 5.1.37).

Solidified HLW from the reprocessing cycle, which contains nearly all of the fission products and very little plutonium, could conceivably be a target of theft for a subsequent threat of dispersal of the radioactive material. However, the handling problems during attempted theft are as formidable for HLW as for spent fuel. Heavy shielding and special equipment are required to avoid serious radiation exposure. These factors make HLW relatively unattractive for theft for any purpose, regardless of the form.

The TRU wastes would also be processed or treated in vital areas until they have been concentrated and/or packaged so they can be transported and stored without hazard. After packaging, the low radiation items may be stored onsite in protected, access controlled areas. The materials in packaged form contain only small amounts of fissionable material, and are unattractive targets for theft. Sabotage would require access to the storage location in the plant. If sabotage is successful, the facility may be damaged and the site contaminated with radioactive waste. The contamination is expected to be contained with little or no public exposure because of the plant location, site layout, and safety features.

The principal products of the example dissolver off-gas treatment facilities are the radionuclides krypton-85, carbon-14, and iodine-129. The krypton will be concentrated and stored as a compressed gas in cylinders and the carbon-14 and iodine-129 will be adsorbed and packaged as calcium carbonate and silver zeolite beds, respectively.

Krypton-85, a chemically inert gas, in the packaged form would be a concentrated radioactive source. The dose rate at the surface of an unshielded cylinder would be about 700 R/hr when filled at the treatment plant. Remote operation in shielded storage areas will be required to process krypton, thus reducing the availability of this waste form and making the cylinders relatively inaccessible targets. In case of a release, the material rapidly disperses and the threat to the health of the general public is insignificant. However, a cylinder rupture outside the facility would probably result in serious exposure to nearby operating personnel. The massive shielding required during transport provides protection against sabotage.

Neither carbon-14 packaged as CaCO_3 nor iodine-129 packaged as a spent silver zeolite bed are attractive targets for theft and eventual dispersal, or for deliberate dispersal onsite by sabotage. In these forms the carbon and iodine are nonvolatile and nonhazardous in the amounts handled or treated in the facility, and too low in concentration to be a health hazard to the public if released onsite as a result of sabotage.

4.10.2.2 Safeguards Requirements for Storage of Reprocessing Cycle Wastes

During the period before ultimate disposal, solidified HLW may be stored in water basins or in surface facilities in sealed casks. Although the waste is not a source of fissionable material, physical protection during storage must be provided to deter and prevent theft or sabotage. The rationale for either theft or sabotage may be to disperse or threaten to disperse the radioactive contents of the casks or storage facilities.

The physical protection requirements for storage of encapsulated solidified HLW in water basins are expected to be the same as those for storage of spent fuel. If the waste were stored at a reprocessing plant, the facility would be a vital area and its physical protection is described in Section 4.10.2.1. If the waste were stored at a separate water basin facility, the safeguards evaluation for spent fuel storage, described in Section 4.10.1.2, apply. The risk associated with the possible sabotage of solidified HLW in water basin storage is probably less than for spent fuel.

If HLW is stored in a sealed-cask storage facility, the facility would be protected against unauthorized entry, forced intrusion, and sabotage. In such a facility the waste canisters are not readily accessible because:

- Remote operation is required to handle canisters.
- Massive biological shielding is required to attenuate canister radiation.
- Facility design features that protect against severe natural occurrences minimize accessibility of the unloading/handling areas.^(a)

The consequences to the public even if a sabotage effort should succeed are, however, expected to be very small. If a canister of waste is ruptured by explosives, the dispersed radioactive material should be confined largely to the storage area because the material is in a solid form and not dispersible except to the extent that pulverization occurs from the explosion energy. Safety analyses of an accidental rupture of a HLW canister inside the building showed that the release of radioactive material would be slight and the public exposure negligible (See DOE/ET-0028, p. 5.4.17).

Packaged TRU wastes are not attractive targets for theft or sabotage because of the low quantities of plutonium and the variable amounts of fission products present. The wastes contain radioactive materials in concentrations several orders of magnitude below those in spent fuel. Much of it would be packaged in 55-gal drums or large boxes. The variations in fission product content will result in surface dose rates that are expected to vary from below 0.2 R/hr to above 10 R/hr.

A sabotage threat will create concerns over radioactive releases. While sabotage may potentially result in some releases to the atmosphere, the amounts released would result in no significant health threats to the public. If a sabotage act causes a bitumen fire, about 10 grams of the fixed waste may be released to the cell, vault or burial crypt atmosphere; lesser amounts would be released to the environment. While an attempted sabotage of TRU waste storage that results in a fire could be a serious incident, the consequences to the public would be small.

The overall physical security required at sites containing TRU wastes protects the public from willful misuse of this waste.

A krypton storage facility will probably be located adjacent to the reprocessing plant. Physical protection of transportation and the storage facility to deter and prevent

(a) Only conceptual plans for such a facility have been prepared. The actual design will involve detailed safety and safeguards analyses.

intrusion or sabotage would be required. The dose rate at the surface of an unshielded krypton cylinder would be about 700 R/hr when received from the reprocessing plant. A remote and shielded storage area will be required for storage, thus reducing the availability of this waste form and making the cylinders relatively inaccessible targets.

It is possible for acts of sabotage to rupture a cylinder of krypton during the receiving or internal transfer operations. The consequences to the public from such acts, however, would be small because the storage buildings are designed to allow the release of krypton through high stacks only. Approximately 104 KCi of krypton-85 might be released over a half-hour period. Such a release could result in significant exposure of workers in the vicinity of the rupture.

Successful sabotage of the krypton storage cell does not appear credible. The cell walls, at least two feet thick, are built of reinforced concrete. However, if the walls are breached by an act of sabotage and krypton is released at ground level, the consequences to workers in the immediate area could be serious but the consequences to the public would be small.

4.10.2.3 Safeguards Requirements for Transport of Reprocessing Cycle Wastes

Solidified high-level wastes will be shipped in casks designed specifically for this purpose. A shipment of HLW will contain more fission product activity but less than 1% of the plutonium included in a shipment of spent fuel. Physical protection requirements for shipments of solidified high-level wastes have not been established by the regulatory agencies. The actual level of physical protection required for shipments of solidified high-level wastes will likely be based on the experience of successful shipments of spent fuel.

Shipping casks as currently conceived, with designs based on the cask criteria for shipping spent fuel, offer significant protection against assault and attempted removal of the contents. A cask would weigh about 90 metric tons, with a special cask cover weighing about one metric ton and requiring special equipment to remove. The cask would be resistant to small arms fire. Explosives in sophisticated designs and arrangements could penetrate a cask. However, the consequences of penetration of a cask would be a minor release of radioactive material at the site (DOE-ET-0029, Vol. 2, p. 8.1.5).

The packaged TRU wastes would be relatively unattractive to an adversary because of its high and varied dilution of radioactive materials. No container or single shipment would contain more than 40 grams of plutonium, and the material, when immobilized in concrete or bitumen or in some other non-dispersible form, would not be a threat to the public as a dispersible radioactive contaminant. If sabotage of a shipment occurs, the release of radioactive materials even under severe conditions is expected to be small (DOE-ET-0029, Vol. 2, p. 8.1.5).

REFERENCES FOR SECTION 4.10

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CHAPTER 5

GEOLOGIC DISPOSAL

In this chapter, the concept of a conventionally mined deep underground repository for disposal of spent fuel and/or fuel reprocessing wastes is described. The status of the technology is described as are uncertainties that require resolution and additional information that would improve confidence in the concept. A description of a conceptual repository for spent fuel or for fuel reprocessing wastes is given. An analysis is presented of the environmental impacts associated with construction and operation of repositories in representative media. Several types of failures of repositories in the long term have been hypothesized to assess societal risk. A description of dollar costs of repositories is also presented. The concern for safeguards is reviewed. Finally, the environmental impacts are summarized in terms of the irreversible and irretrievable commitment of resources and in terms of unavoidable adverse impacts.

5.1 DESCRIPTION OF THE GEOLOGIC DISPOSAL CONCEPT

Geologic disposal of radioactive wastes, as used in this Statement, is the disposal of radioactive wastes in conventionally mined repositories deep within the geologic formations of the earth. Included is the concept of multiple barriers to provide a series of independent barriers to the release of radionuclides to the biosphere.

The multiple barriers that could contain nuclear waste in deep mined repositories fall into two categories: 1) geologic or natural barriers and 2) engineered barriers. Geologic barriers are expected to provide isolation of the waste for at least 10,000 years after the waste is emplaced in a repository and probably will provide isolation for millennia thereafter. Engineered barriers are those designed to assure total containment of the waste within the disposal package during an initial period during which most of the intermediate-lived fission products decay. This time period might be as long as 1000 years in which case the radiation levels and heat generation rates of the total waste would drop by factors of approximately 1,000 and 100, respectively. Engineered barriers must be designed to withstand the more severe radiation and thermal conditions encountered initially.

Two important components of the geologic barrier to be considered in siting are the host rock itself and the geologic surroundings. Properly chosen rock structures provide physical and chemical properties that contribute to repository strength. Sufficient repository depth and lateral extent of the rock mass contribute to the isolation capability of the repository. Tectonic stability and a noncommunicating hydrologic regime combine with rock properties to maintain repository strength and isolation integrity. The geologic barriers can be selected through the site-selection process to provide a stable long-term environment for the waste that is not likely to be disturbed by natural events or human activities.

This section provides an overview of general considerations in the design and location of geologic repositories. Additional details including references to specific studies in the literature are given in Appendix B of Volume 2. Details of both engineered and natural barriers to waste release are also presented.

5.1.1 Factors Relevant to Geologic Disposal

Six factors relevant to geologic disposal are:

1. Depth of repository below the land surface. Presently it is assumed that a range of from 600 to 1,000 m of earth material will exist between the repository and the land surface. This will provide a barrier between the waste and the biosphere and protect the repository from human activities. Dimensions of the host rock are also considered so that the repository will be buffered by rock material laterally and below as well as above it. An artist's conception of a repository is shown together with more familiar structures in Figure 5.1.1.
2. Properties of the host rock. The physical, chemical, and thermal properties of the host rock determine the rock's capability to isolate and contain the waste and reduce unwanted interactions between the rock and waste. These possible interactions include radiation effects on the rock and chemical and physicochemical interactions. Important rock characteristics include strength, permeability, thermal conductivity and expansion, and radiation resistance.

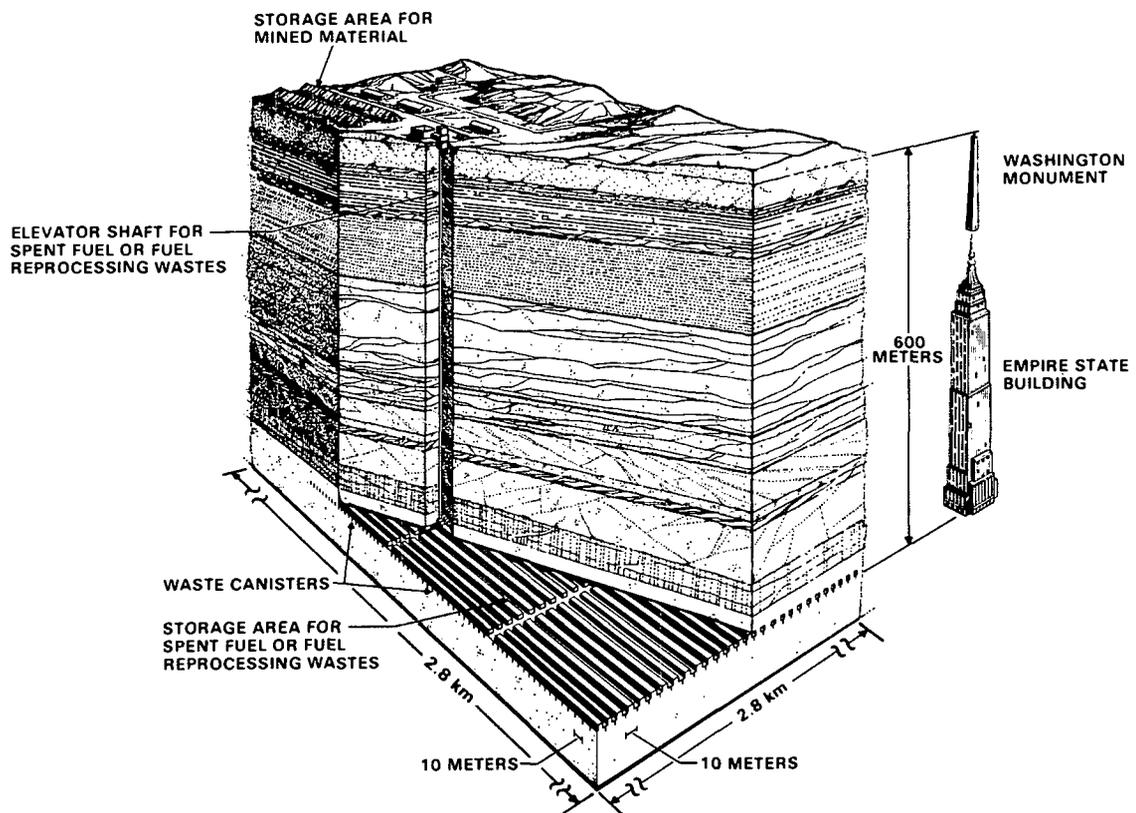


FIGURE 5.1.1. Deep Underground Geologic Waste Repository

3. Tectonic stability of the repository area and region. Proper consideration of this important factor will reduce the likelihood of deformation or disruption of the host rock and thus increase the probability of repository integrity.
4. Hydrologic regime (i.e., surface-water and ground-water considerations). This is important because the existence of connected water channels could provide potential pathways for waste transport away from the repository.
5. Resource potential of the repository site and area. A low resource potential is desirable to avoid loss of any economic resource by the repository existence and to reduce the likelihood of future exploration activities for resource recovery.
6. The multibarrier safety feature. This combines the redundant isolation features provided by the rock properties, the geologic setting, and engineered barriers to give overall added confidence that the waste will remain isolated.

These six factors are discussed in the following sections.

5.1.1.1 Disposal Media Properties

Four geologic media are examined in this Statement to illustrate a range of rock properties for a radioactive waste repository: salt deposits (bedded and dome), granite, shale and basalt. All four rock types possess properties that are favorable for waste isolation. These, as well as some unfavorable characteristics, are discussed in the following pages.

For the purpose of this Statement, the physical properties of a disposal medium describe the characteristics of both the host rock and surrounding rock mass. The disposal rock material is characterized in terms of its texture, i.e., the size, shape, and arrangement of the component crystal grains. Texture is a consideration in the assessment of a medium's behavior under stress and heat, and its hydrologic flow potential.

Rock mass structures include the discontinuities of bedding and joints. Bedding refers to variations in texture because of changes in the sedimentation process by which the rock was formed. It may be present in both sedimentary and metamorphic rocks. Joints are fractures along which little or no displacement of the rocks has occurred. They are generally formed by extensional release of confining earth pressures. Descriptive features of these discontinuities include orientation, width, spacing, filling material, waviness, and extent (length). The potential for the transport of waste material correlates with the number and extent of host rock discontinuities.

The rock properties of principal interest for waste disposal are those related to strength, stress-strain, thermal, and hydrologic characteristics. These properties and characteristics are discussed and presented in tabular form in Appendix B. For comparative purposes index properties defined as unit weight and natural moisture content are included in the tabulation.

Substantial strength is desirable for engineering design of subsurface repository facilities, especially in maintaining tunnel integrity. Strength properties provide the durability or resistance of a material to processes such as erosion and weathering and

breakdown into component minerals. In general, the greater the strength, the greater the ability to resist weathering. Parameters representative of strength include cohesion or friction angle, uniaxial compressive strength, and tensile strength.

Stress-strain properties indicate the deformation characteristics that a material will exhibit under stress. Parameters that describe the nature of the deformation of a disposal medium include Young's modulus, Poisson's ratio, bulk modulus, and shear modulus. These parameters are significant in the analysis of an earth material's strength, durability, and use properties, such as mineability, for isolation. The desirability of (or, trade-offs between) a highly deformable medium versus a rigid disposal medium for isolation purposes is unresolved. The ability of an earth material to deform and seal discontinuities to fluid flow is desirable. Conversely, a rigid earth material is important to the stability of the repository tunnel opening.

Thermal properties indicate an earth material's ability to absorb and conduct heat away from radioactive waste. Knowledge of these properties will allow the evaluation of the effect of the heat upon the integrity of the disposal medium. Pertinent thermal parameters are coefficient of linear thermal expansion, heat capacity, and thermal conductivity. Heat can physically alter an earth material by causing expansion, which, in a confined disposal medium, can jeopardize isolation. For example, too much expansion of the rock might fracture the overburden above the repository. The degree of expansion is dependent on the ability of a host rock to dissipate heat and dependent on the amount of expansion for a given temperature change.

Hydrologic properties are essential to assessing the potential for fluid flow. They are evaluated by the parameters of permeability, hydraulic gradient, and porosity. Restriction of transport of radionuclides requires as low a permeability as possible.

A host rock is an aggregate, composed of one or more naturally occurring minerals and chemical compounds. The constituents provide the chemicals for potential reactions of the host rock with the waste material. These possible reactions may increase isolation by precipitation of insoluble materials or decrease it by converting radioactive waste into soluble compounds. Possible chemical reactions among disposal media, intergranular fluids, and waste must be defined and evaluated for their effect on isolation.

Disposal media of salt, granite, shale, and basalt are examined here and represent only a selected sample of candidate host rock types. Other host rock types may also meet the requirements for media properties and distribution. Additional media can be grouped as having properties similar to those of the example media. Associated disposal media are grouped as 1) salt: anhydrite, gypsum; 2) granite: general crystalline rock, granodiorite, peridotite, gneiss, syenite; 3) shale: general argillaceous rock, carbonate; and 4) basalt: gabbro and some tuffs.

5.1.1.2 Generic Basis for Repository Site Selection

This section presents the generic basis for repository site selection and the design of the repository. Characteristics most desirable for site selection and how they relate to design are discussed. Criteria necessary for development of siting criteria and repository waste form design are presented.

The most important site-selection factors can be derived from the six geologic considerations given in Section 5.1.1.1. In general, the most important factors are the hydrologic regime, the tectonic regime, the multibarrier concept, and the thermal, physical and geochemical properties of the host rock. For any particular location, site-specific considerations peculiar to that site might be different and would take precedence.

The site-selection process will proceed in stages as described below. Program scientists will select regions, areas, and sites, in that order, by their meeting defined requirements. Each stage of the site-selection process will add to the geologic information available for the preceding stage and will better define uncertainties. Therefore, the site-selection process will yield progressively more significant data; that is, each phase of the process will further characterize site-specific considerations, thus reducing uncertainties.

The following criteria are suggested for repository site selection to assure that the natural barriers function as planned:

1. The repository site shall be located in a geologic environment with geometry adequate for repository placement.
2. The repository site shall have geologic characteristics compatible with waste isolation.
3. The repository site shall have subsurface hydrologic and geochemical characteristics compatible with waste isolation.
4. The repository site shall be located so that the surficial hydrologic system, both during anticipated climatic cycles and during extreme natural phenomena, shall not cause unacceptable adverse impact on repository performance.
5. The repository site shall be located in a geologic setting that is known to have been stable or free from major disturbances such as faulting, deformation and volcanic activity for long time periods.
6. The repository site shall be located in an area that does not contain desirable or needed mineral resources, or to the extent presently determinable, resources that may become valuable in the future.

Regional studies of stratigraphy and structural geology will be conducted to aid in site selection. Stratigraphy is the general characterization of the sequence of rock types both vertically and laterally. Structural geologic studies determine orientation of the rock units in space, direction of dip, configuration of folds, and the characteristics and attitudes of faults, joints, and other discontinuities. Adequate description of the

geologic setting may require extensive geologic mapping, some field exploration, and remote sensing surveys, especially in areas that are not yet thoroughly studied.

5.1.1.3 Generic Basis for Repository Design

Several conceptual designs for repositories have been proposed. The surface structures of a repository do not present unique engineering problems. The typical conceptual design of the underground portion of a repository consists of numerous excavated storage rooms (at one or more levels) interconnected by tunnels which serve as transportation and ventilation corridors. The undisturbed rock masses that separate the storage rooms are called pillars. Boreholes will be drilled into the floors (and possibly walls) of the rooms. The waste will be placed in these boreholes. The repository levels are reached from the surface handling facilities through vertical shafts.

The integrity of a repository will depend largely on the state of stress level in the rock, the ground-water flow, the strength of the rock, heating and radiation effects from the wastes, and the layout of the excavations and the disposition of the waste within them. A large body of pertinent data exists which presents and analyzes each of the above factors. The results indicate that there are no fundamental geological or mechanical reasons why excavated repositories should not be used at suitable sites in rock.

The cost of excavating the repository and the cost of rock support systems depend on several interrelated geologic factors: rock strength, rock fractures, rock hardness, rock permeability, rock heating by decay of radioactive nuclides, the state of rock stress, the depth of waste placement, and others. The extent to which these factors influence cost is difficult to determine in advance of construction; unforeseen rock conditions are often encountered in conventional mining operations and in some cases can significantly change the design and the predicted cost. Cost estimates for geologic repositories are given in Section 5.6 of this Statement.

5.1.2 Engineered Barriers

The multiple barrier concept of waste isolation and containment includes both natural or geologic and engineered barriers. Various aspects of engineered barriers are discussed in this section.

5.1.2.1 Engineered Barriers--Waste Package System

The term "waste package" as used in this Statement includes everything that is placed in the waste emplacement hole, e.g., the solidified waste form, canister, overpack, filler and backfill materials, and hole sleeve. The function of the waste package is to:

- Contain the waste for periods sufficient to allow most of the fission products to decay to very low levels.
- Limit the rate of release of radionuclides to the near-field (within the repository proper, see p. K.4) host rock system.

- Limit access of water to the waste and thereby prevent or minimize waste/rock/leachant interactions.

The functions and materials use for waste package components can be tailored to specific site needs and environmental factors. A conceptual representation of a waste package system is given in Figure 5.1.2. Not all of the components shown here would necessarily be used in all circumstances; the figure illustrates the different kinds of barriers that can be engineered into the waste package. The overlying principle is to design into the package as much redundancy as required by characteristics of the waste to be contained and the characteristics of the natural geologic system.

Waste Package Functions

One may envision how the waste package is designed to function by considering the case of ground water intruding into the repository. A basis for repository site selection will be remoteness from aquifers, so the amount of water should be small. If water intrudes into the repository it would first encounter the backfill, which can be designed to be relatively impermeable to water by reason of its physical and chemical properties. Any water passing through the backfill would encounter a sleeve or overpack, or both, made of corrosion-resistant materials. As a further redundant measure, the canister itself would act as a physical barrier. If all these sequential barriers to water influx were to fail, the waste form itself would be a barrier because of its low solubility and resistance to leaching. If some nuclides were mobilized by ground water, they then would have to travel through damaged package components until they reached the backfill again. The backfill may then function as a sorptive barrier to retard or minimize transport of selected nuclides. Thus, the total waste package system can be designed to minimize the nuclide inventory entering the natural system, by chemically and physically limiting nuclide mobility and by delaying

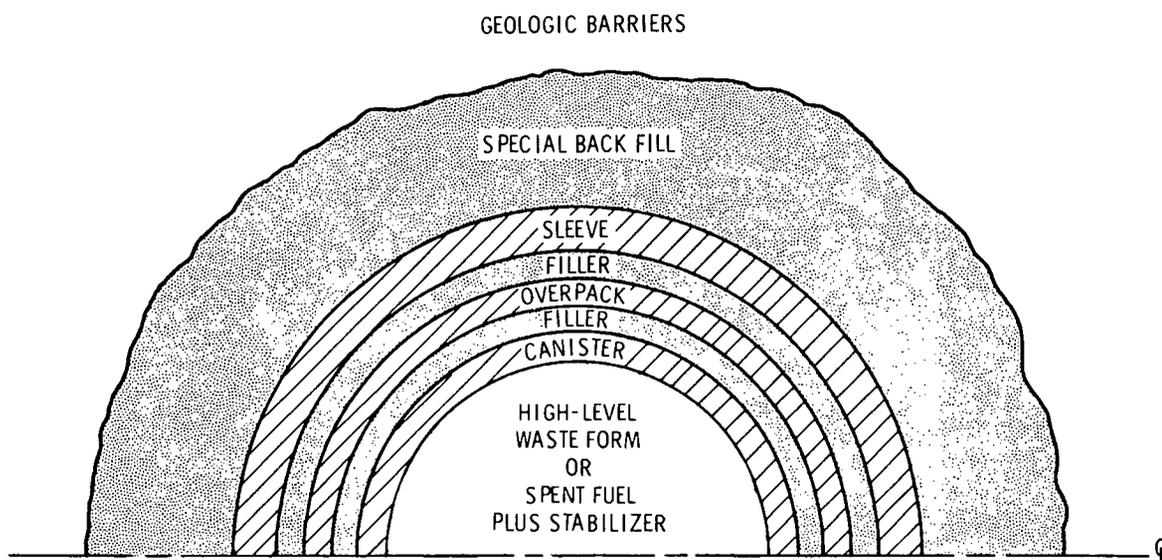


FIGURE 5.1.2. Conceptual Waste Package

releases so that substantial nuclide decay occurs before entering the geologic system where the natural barriers would prevent or delay releases to the biosphere.

5.1.2.2 Waste Packages Components

Components of a generalized waste package were shown in Figure 5.1.2. The following discussion addresses each component separately; however, it is the performance of the entire system of components taken as a whole that is of most importance in the final analysis.

Waste Form

The waste forms include all radioactive materials that may potentially be sent to deep geologic repositories, and are divided into three major categories: spent fuel, high-level waste and TRU waste forms, which are described in more detail in Sections 4.3.2, 4.3.3, and 4.3.4, respectively. The current primary emphasis on waste package design is for spent fuel and for HLW, the reference waste forms considered throughout the following discussion. Due to their high radiation levels and heat generation, spent fuel and HLW place the most stringent requirements on the waste package. However, when most of the fission products have decayed (after a few hundred years), the properties of the TRU waste become dominant.

The waste form is an inert solid designed to be chemically, thermally and radiolytically stable. The waste form itself is the first containment barrier for the waste.

Canister

The canister provides physical containment for the waste forms and thus isolates the waste from near-field surroundings. The extent to which the canister can delay or minimize waste-water interactions is important. Moreover, the canister is expected to provide physical protection during interim storage, transportation, handling, emplacement, and any waste retrieval operations that may be required. The canister material chosen must be compatible with the waste form. The ductility, weldability and impact resistance of metals make them primary candidates as canister materials.

High-level waste forms will generally fill the canister 80 to 90% full. The remaining space will be occupied by air. Stabilizer materials are being considered for use in spent fuel canisters. Gaseous stabilizers, such as helium, have been considered from the standpoint of providing a heat transfer medium without causing chemical or mechanical attack on the spent fuel/cladding assembly or the canister. Particulate or solid stabilizers, such as lead, glass, clay, or sand, can provide additional functions, including maintaining the position of the spent fuel within the canister; preventing canister collapse under lithostatic pressures; acting as a corrosion resistant protective barrier; improving heat transfer; increasing radiation attenuation; and enhancing nuclide sorption.

Overpack

The overpack is similar in principle to the canister. An overpack offers several options to the package designer: it may function as a redundant canister, applied (if necessary) for all stages of package handling, transportation, and emplacement; it can exhibit corrosion or mechanical properties superior to those of the primary canister,

thereby providing all, or a major part, of the resistance to the environment required by the package longevity criterion; it can provide a degree of uniformity to a variety of canister types, applied at the repository to accommodate acceptance criteria. The canister and overpack together can be referred to as the "container."

Overpacks for use in the repository are designed especially for chemical durability, with less emphasis on properties such as impact resistance that are mainly important during handling and transportation. Thus, a wide range of materials, in addition to metals, are being studied. These include various ceramic materials, graphite and carbon materials, a wide variety of glasses and specially selected cements.

Emplacement Hole Backfill

Backfill materials are designed to fulfill one or more of several functions:

- Sorbing the limited amount of water that may be present in a repository rock, e.g., from brine inclusion migration in salt.
- Impeding the movement of intruding ground water to and from the waste package.
- Selectively sorbing radioisotopes from ground water in the event of the canister breach.
- Modifying ground-water chemistry and composition (e.g., pH, Eh, etc.) to reduce corrosion rates or minimize waste form leaching.
- Providing mechanical relief by accommodating stresses on the waste package induced by rock movement.
- Serving as a heat transfer medium.

Several layers of filler or backfill material can be utilized, if desired, as shown in Figure 5.1.2; thus, different materials specially designed for specific purposes can be included for optimum functioning of the overall waste package system. Most of the filler or backfill materials being considered are naturally occurring clays, sand or crushed rock that are readily available in large quantities.

In addition to backfill in the emplacement holes, backfill material is also placed in rooms and corridors when the repository is closed. The room and corridor backfill, depending upon the material and method of emplacement, can perform the same functions described for the hole backfill. The degree of structural support provided may be important in preserving repository integrity by limiting the subsidence of room and corridor ceilings. The permeability and porosity of the backfill material may affect the amount of water entering the repository and the time it takes for the repository to become saturated.

Mechanically emplaced crushed rock is used for backfilling the conceptual repository described in this Statement. The use of an engineered sorption barrier as backfill is discussed in Appendix K. Other backfill materials and methods of emplacement are discussed in NUREG/CR-0496 (NRC 1979).

Hole Sleeve

The function of hole sleeves is to maintain open emplacement holes in the repository floor for easy package insertion and retrieval. This may be important if the geologic medium is plastic, e.g., salt or certain shales. In some cases the sleeves could function simply as barriers that, because of their size and bulk, are more easily constructed in situ than transported and emplaced with the waste canisters. Examples of sleeve configurations include cast iron caissons, massive shells of special cements cast in place, or impervious graphite vessels specially bedded in the surrounding rock.

5.1.2.3 Waste Package Development and Assessment

Although most of the ideas incorporated in the multibarriered waste package just described are not new, wide-spread acceptance of the waste package concept is a relatively recent development. A study done in Sweden between 1976 and 1978 did a great deal to promote acceptance of the concept.

The Swedish Approach to Waste Package Design

In April 1977 the Swedish Parliament passed a law which stipulated that new nuclear power units could not be put into operation unless the owners were able to show that the waste problem was solved in a completely safe way. In anticipation of Parliament's action the Swedish power industry formed the Nuclear Fuel Safety Project (KBS) in December 1976 to prepare a response to the government (KBS 1978). A primary objective of the KBS project was to demonstrate how high-level waste or spent fuel can be handled and finally isolated. The study met this primary objective, and although the results were directed to the specific needs of one country and assumed a repository located in granite since that type of rock is widely available in Sweden, the conclusions about the expected performance of the waste packages can have a wider application.

The KBS decided to place reliance on containment for periods of 1,000 yr and 10,000 yr for HLW and spent fuel, respectively; thus, design of the waste package received heavy emphasis. More durable containment for the spent fuel was sought because it produces significant amounts of heat for a longer time than does HLW.

In the proposed Swedish waste management scheme for HLW, the fuel is reprocessed 2 to 10 yr after it is taken from the reactor (KBS 1978, pp. 30-34). The HLW is then vitrified and is placed in cylindrical stainless steel canisters that are stored at the reprocessing plant for at least 10 yr. After this initial storage period, the canisters are shipped to an underground air-cooled dry storage facility in Sweden, where they remain for about 30 yr. Then the packages are prepared for disposal by encapsulation in 6-mm-thick titanium overpacks. To reduce the intensity of radiation emanating from the packages and hence the radiolytic decomposition of the ground water eventually expected to surround the package, a 10-cm-thick layer of lead is placed between the steel canister and the titanium overpack. The packages, now ready for disposal, would be placed in holes approximately 1 m in diameter and 5 m deep in the floors of tunnels in a granite repository approximately 500 m below the surface of the ground. Backfill consisting of a mixture of quartz sand and bentonite is

packed around each package. After all holes are filled, the entire tunnel system is filled with a mixture of sand and bentonite similar to that used in the storage holes.

A "reference group" made up of members of the Swedish Corrosion Research Institute concluded that the stainless steel/lead/titanium composite canister could be expected to remain intact for 500 to 1000 yr, even when very pessimistic assumptions were used (KBS 1978, p. 110).

At least two waste package designs appear capable of achieving the longer life sought for spent fuel disposal. In one design the spent fuel is encapsulated, after about 40 yr of interim storage, in copper canisters 77 cm in diameter with walls 20 cm thick (KBS 1978). The other design utilizes a synthetic corundum (Al_2O_3) canister. A feasibility study has shown that it is possible to manufacture such canisters using hot isostatic pressing. Each canister would have an interior diameter of 0.3 m, a 100-mm-thick wall, and be about 3 m long.

Although the Swedish waste disposal packages may be more complex than some packages now under study, they have served to increase our understanding of long-term package performance.

5.1.2.4 Current Status of Waste Package Development in U.S.

Extensive testing and development studies on various individual barrier components of the waste package system, under expected conditions of geologic isolation, have been in progress for several years. These studies are being conducted in industrial and national laboratories and in universities. While most of the studies are not complete, data and results generated during the past few years indicate that components of the waste package system, individually and in combination, can prevent or minimize release of radionuclides outside of the repository by functioning as effective chemical and physical barriers (Katayama 1979, Ross and Mendel 1979, Braithwaite and Molecke 1978, McCarthy et al. 1979, Magnani and Braithwaite 1978 and Nowak 1979).

Through laboratory materials performance evaluation under realistic repository environmental conditions and accelerated aging tests, a number of waste package candidate materials are being selected. Following laboratory testing, nonradioactive bench-scale experiments and radioactive hot cell experiments are planned. These tests employ small-scale mockups of complex systems or groups of system components to investigate the influence of components upon each other. For example, leaching/corrosion studies utilizing a scaled down canister of an actual waste form with rocks and ground waters are in progress (ONWI-9(4)).

The logical culmination of a series of studies investigating waste package material performance and qualification is a field test specific to each repository rock type which involves all components of the waste package. The extent of field testing will be determined from the analysis of earlier results. Various aspects of required laboratory and field tests have been described by the U.S. Geological Survey and the DOE in the Earth Science Technical Plan (DOE/USGS 1980). A Waste Package Design, Development, and Test Plan is being formulated to direct development efforts in an effective and timely manner. An

integral part of this plan is the development of coordination among and standards to be followed by researchers and waste management program entities with respect to testing procedures and materials certification. Review and integration entities are defined to include a Materials Steering Committee, a Materials Review Board, a Materials Characterization Center, and an Independent Measurement Standards Laboratory (Hindman 1980). This organization and plan will help assure that waste package design, development and testing programs will produce suitable packages that meet established requirements.

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5.2 STATUS OF TECHNOLOGY AND R&D

Research and development is underway to address the data needs of waste isolation identified in this Statement. In conducting R&D for waste isolation, a technically conservative systems approach is being used, with emphasis on scientific peer review of the activities along with public review, such as the public comment activities of this Statement.

An important document supporting DOE's R&D effort is the Earth Science Technical Plan (ESTP) for Disposal of Radioactive Waste in a Mined Repository (DOE/USGS 1980). The ESTP was prepared by a group consisting of scientists from USGS, DOE and DOE contractors. This group has comprehensively reviewed R&D to define work that may improve the reliability of isolating nuclear wastes in a mined geologic repository, and has recommended programmatic activities. The ESTP describes R&D programs sponsored by DOE and the U.S. Geological Survey. The work in progress involves 76 R&D contractors (including 20 universities and 7 national laboratories). While the key work in progress is discussed in the paragraphs below, the reader is referred to the ESTP to gain more complete perspective on the ongoing R&D activity. Parallel studies sponsored by NRC, EPA and the utility industry are in progress in the United States and in foreign countries (particularly Sweden, Federal Republic of Germany, France, Great Britain, Japan and Russia).

The following sections provide a general discussion of the current status of technology and the R&D activity and requirements for the geologic site selection, waste package, and repository system.

5.2.1 Geologic Site Selection^(a)

Geologic site selection involves characterizing promising areas of the United States as possible locations for repository facilities for radioactive waste (see also Section 2.3). During site selection or qualification, certain factors or criteria necessary for adequate performance of the natural system must be considered. Such factors or requirements are summarized in the "NWTS Criteria for the Geologic Disposal of Nuclear Wastes: Site - Qualification Criteria (ONWI-33(2), 1980)." These requirements are being used by DOE to guide its site selection or qualification activities until such time as formal licensing criteria are adopted by the Nuclear Regulatory Commission and the Environmental Protection Agency.

Much of the data base for site selection is available. These data include topography, records of seismic activity and volcanism, hydrology, and presence of the natural resources. Other data, including depth to a potential emplacement zone, areal extent of rock type, attitude (dip, inclination), and the nature of the contiguous formations are developed at specific sites. Ground water, as the principal agent for transport of radioactive waste to the biosphere, has received intensive study and research over the past decade. The principles that govern its occurrence, movement and related rates of supply and usage are well established. While major aquifers and their distribution and properties are known, additional study using accepted techniques can define regional and local flow systems adequately.

(a) Section 2.3 describes the present National Site Characterization and Selection Plan.

Specific topics elaborating on site selection criteria and the supporting R&D addressing these matters are discussed below. Supporting R&D projects are listed by organization in Appendix L of Volume 2.

5.2.1.1 Methods for Regional Geologic Studies

Geologic studies will identify, for a specific region, area and site, the current state of stability and the geologic processes which have acted in the past. Based on this information along with repository design, the projection and probability for the future stability of the specific site will be estimated (see Appendix L).

General geologic conditions in the United States are well known and have been extensively described (Geologic Society of America, current listings). Exploration for mineral resources--notably oil, gas, coal, and metals--by private industry provides much information about sub-surface geologic conditions, in many instances to depths approaching 10,000 m (Am. Assoc. of Petroleum Geologists, current listing). The construction of nuclear reactors, which must meet stringent licensing requirements, has resulted in detailed geologic evaluations of areas in the Eastern, Midwestern, and Far Western United States (FUGRO, Inc., 1977). Moreover, various universities have developed as centers of detailed geologic information on specific subjects. The accumulated knowledge is sufficient to identify areas in the United States that meet many of the requirements (Section 5.2.1) for radioactive waste repositories.

5.2.1.2 Methods for Site Analysis

In general, geologic studies are the mechanism by which available data about the sub-surface environment are synthesized and coordinated to assess whether the stratigraphic and structural settings of a proposed site are suitable for a waste repository. Remote sensing and geophysical studies are conducted to support this activity. Geologic interpretations are the basis for defining models by which the hydrologic, geologic, geochemical, thermal, and mechanical characteristics of a repository are assessed.

Geophysical Surveys

Geophysical surveys are an integral part of site selection and characterization studies. Many of the geophysical techniques utilized by the petroleum and mineral industries have been applied to the search for geologic repositories. The broad categories of exploration geophysics summarized in this subsection are gravity, magnetic, electrical, and seismic methods. In addition, well logging and borehole geophysics are discussed.

There currently exists a wide variety of geophysical techniques available for site selection and characterization. Geophysical surveys are a well established part of exploration prospecting and proper evaluation can provide extensive information about subsurface geologic conditions. Such surveys are especially valuable in repository investigations because they permit investigation of subsurface conditions without extensive drilling.

Gravity analysis can detect small variations in the earth's gravity field (Dobrin 1960). The variations of principal interest to repository siting result from lateral variations in subsurface rock density. Density variations may result from deformed strata, faults, igneous intrusives, diapirs, breccia pipes, or lithologic changes.

Magnetic methods detect variations in the earth's magnetic field (Dobrin 1960). The magnetic variations (anomalies) of interest to site studies are due to lateral changes in mineral content (especially magnetite) or to variations in the remnant magnetism of igneous rocks. Subsurface structures like anticlines or faults can be detected if they result in lateral changes of the above properties (Fabiano 1976).

A variety of electrical methods (Dobrin 1960 and Keller 1966) are used in geophysical exploration; all depend upon detecting variations in the electrical resistivity of the media through which a current flows. Subsurface resistivity is highly variable and strongly influenced by the amount and the nature of fluids in the rocks. For this reason, such hydrologic features as dissolution of salt, ground-water tables, and porosity variations are particularly amenable to electrical prospecting methods.

Seismic exploration methods are perhaps the most useful geophysical tools for obtaining accurate representations of the subsurface geology at individual sites (Dobrin 1960). They rely on the reflection or refraction of seismic (acoustic) signals due to contrasts in velocity or acoustic impedance (the product of seismic velocity and rock density). Acoustic signals are usually introduced into the earth by explosive sources or vibrating or impacting masses. Seismic reflection surveys are particularly useful in mapping the attitude and continuity (or lack thereof) of subsurface rock beds. Other methods and equipment utilized for seismic reflection can be selected for the specific site and parameters (i.e., depths, dimensions) of interest. These parameters are often defined to provide information from depths of more than 1,000 meters (Vail et al. 1978). Special field parameters and techniques (high-resolution seismic) are available to explore accurately the shallower depths of interest for repositories.

Geophysical logs in well bores are a powerful tool for correlating and interpreting subsurface geologic conditions, including the condition and fluid content of subsurface rocks (Dobrin 1960). They supplement cores and rock samples and furnish a vertically continuous record of certain physical properties for each borehole. Many types of logs are used. Focused resistivity logs provide a reliable measure of in-situ rock and fluid characteristics. Microresistivity logs measure the properties of small volumes of rock just behind the borehole wall and thus permit the boundaries of permeable and/or electrically resistive formations to be sharply defined. Gamma-ray logs indicate the clay content of various formations and are valuable in making lithologic interpretations in clastic rock sequences. Neutron logs are useful for identifying porous rock strata and rock densities. These logs respond mainly to the hydrogen content of the formation and indicate the presence of water, oil, or hydrogen-bearing minerals. Acoustic logs measure the velocity of sound in rock units and can also help determine the porosity of a formation.

Hydrologic Technology

The role of hydrologic studies in site exploration can be separated into three overlapping areas: 1) two-dimensional characterization of the surface and ground-water systems for the region or hydrologic basin in which the site is located, 2) three-dimensional characterization of ground-water conditions at candidate sites, and 3) the potential effects of the repository, the climate, or other perturbations of the ground-water system.

Because it is believed that hydrologic transport will be the principal mode of translocation of radionuclides, a considerable amount of field and test data will be acquired to assess the hydrologic system. The techniques for obtaining most of the data are currently available; others, including improved techniques for ground-water dating, fracture-flow modeling, and permeability determinations for low permeability rocks, need development (Barr et al, 1978 and Bredehoeft et al. 1978). Hydrologic models combined with geochemical studies are used to estimate the likely composition and concentration of any and all radionuclides at any given point and time relative to a site's regional aquifer system.

Data from hydrologic testing are combined with geologic interpretations of a site and region to produce a detailed three-dimensional model of the near-field (see p. K.4) hydrologic flow system which includes the fracture-flow conditions. This is then integrated with thermal and mechanical models to calculate the near-field disposition of the wastes should they escape containment. The near-field models determine the source terms for regional two-dimensional flow models of a subject hydrologic basin. These regional models are used to calculate the isolation potential of the far-field natural system. Retardation mechanisms (e.g., sorption, precipitation and diffusion into the rock matrix) and radioactive decay chains for the radionuclides will be factored into both near- and far-field models of the isolation system. Conservative assumptions regarding potential changes in the hydrologic system that may be caused by climatic and tectonic changes will be used to develop scenarios for modifying models of present ground-water flow conditions.

Permeability, effective porosity, and rock compressibility can be determined by pump or injection tests in wells at the depth intervals of interest. Hydraulic properties are routinely measured for laboratory specimens of core or other rock samples obtained from the site (Lin 1978). Using appropriately spaced wells, hydraulic communication between them can be established during pump or injection tests (Davis et al. 1966) to provide reliable calculations of in-situ ground-water velocities.

Isotopic dating of ground water (Barr et al. 1978 and Bredehoeft et al. 1978) provides an alternative reference for evaluating calculated velocities. Water can be sampled for dating from selected discharge points and well locations throughout the ground water basin considered likely to be influenced by a repository. Differences in water ages among sampling points are used to calculate natural velocities.

The identification and analysis of hydrologic conditions in nearly impermeable rocks is necessary to establish the degree of impermeability possessed by the host rock unit

(Witherspoon 1977). Pulse injection tests aid in determining permeability in low permeability rocks (Ballou 1979). Moreover, pressure decay curves for gases pressurized at selected borehole intervals can be used to estimate the permeability of the very tight rocks expected at repository horizons. Although present measurement techniques for hydraulic conductivity in nearly impermeable rocks may be in error by up to a few orders of magnitude (Bredehoeft et al. 1978), even the higher, most conservative values indicate that water moves extremely slowly in these rocks.

Hydrologic R&D Studies

For rocks that possess a natural fracture system (e.g., granite, basalts, some shales, limestones, sandstones) the determination of near-field flow mechanisms is also evolving. Because fracture networks are not random, their nature and orientation within the system will be statistically determined. Methods designed to assess fracture effects on hydrologic flow are currently being developed at the Nevada Test Site (Johnstone 1980), the Stripa mine in Sweden (Gale et al. 1979), and the Los Medanos site in New Mexico (Gonzales et al. 1979). The direct determination of hydrologic parameters in fracture networks includes conventional pump testing with multiple-point piezometers, tracer studies, and flow-meter tests performed in wells or subsurface facilities constructed at the repository site or in rock bodies that provide a close analog of site conditions.

Water influx at mines in crystalline rocks is a well-known phenomenon. However, where permeabilities are very low, mine ventilation commonly evaporates and removes most, if not all, of this water (Gale et al. 1979). Thus, the mines are usually "dry," although a small amount of water may continually flow into them. By sealing a room with airtight bulkheads and circulating controlled quantities of warm air, the amount of seepage water can be determined by measuring the humidity and mass of the circulating air. Data on fluid gradients around the sealed-off chamber permit calculations of nearby rock permeabilities. Such an experiment is being performed at the Stripa mine in Sweden (Gale et al. 1979 and Lawrence Berkeley Laboratory 1978).

Site-Specific R&D

The thermal properties of potential host rocks can be measured in the laboratory by accepted methods (Stephens et al. and Jaeger et al. 1979). Standard sized cylindrical specimens are subjected to a controlled thermal power source at one end; increasing temperatures and dimensions are measured either along the axes or along the outside lengths of the specimens. The results are then used to calculate volumetric expansion coefficients and thermal conductivity. The specific heat of a rock is determined by standard calorimetry (Stephens et al.).

Mechanical properties of potential host rocks can also be measured in the laboratory by standard techniques and apparatus (Jaeger et al. 1979); the results are used in preliminary models of the repository's response and to help determine which properties require better definition by field testing (Chan et al. 1980). The compressive strengths of

potential host rocks are determined in accordance with well-accepted methods by observing which states of stress and temperature cause fracturing. Standard tests are also performed to determine the tensile strength of rocks.

Synergistic effects between thermal and mechanical properties are determined for laboratory samples by obtaining data on mechanical response as a function of rock temperature or obtaining thermal conductivity data as a function of rock stress. The effects of the rock's fluid content on specific heat, critical stress, and thermal conductivity are also being investigated.

Sorption capacities are currently determined by passing water doped with radionuclides through the rock and measuring the amounts of radionuclides retained. Transient batch methods for determining sorption are currently being standardized (Brandstetter et al. 1979). Techniques are also being developed to identify mineralogic and molecular affinities for sorbed radionuclides, allowing a better understanding of the materials and mechanisms responsible for the sorption process.

Laboratory tests are being validated by field determinations of thermal, mechanical, and chemical behavior under expected repository conditions. Field tests generally involve single or multiple heat sources emplaced in drill holes with an array of measuring instruments surrounding the heat source. A monitor array can be designed to measure rock temperatures, deformation, water content, chemistry, and rock stresses as a function of time and distance from the heat source.

Regional Geologic Forecasting Studies. Predicted performance of a geologic system has not matured to the point enjoyed by conventional engineering disciplines. Geologic research has largely concentrated on characterizing present-day natural processes and events and on historically reconstructing the distribution, magnitude, and sequence of past events. However, future tectonic activity, including volcanic eruptions, folding, epeirogeny, fault movements, salt diapirism, and seismic activity, need to be predicted to the degree that the likelihood and the consequences of changes in the natural system with regard to containment and isolation can be estimated.

Plotting space-time relationships of past events allows a calculation of past rates and distributions of occurrence for tectonic events (Crowe 1978 and Rogers et al. 1977). The probabilistic extrapolation of these rates into the future must be weighted against deterministic tectonic models such as plate tectonics to determine whether observed space-time distributions are likely to continue or be modified. The geographic scale for which data are compiled is of critical importance and needs to be evaluated. In general, for larger areas, consensus is more readily obtained among earth scientists about tectonic processes. Conversely, averaging probabilistic projections for individual events over large areas decreases their reliability for a given site. Thus, a reasoned interpretation of probabilistic and deterministic approaches is required to assess the likelihood of tectonic events that might disrupt a repository's natural system. This combination of methods is most developed for assessing seismic hazards (Algermissen 1976 and Glass et al. 1978).

Potentially active faults can be deterministically identified from geologic, geophysical, seismic, and natural stress data. Standard earthquake-hazard assessment provides probabilistic estimates of expected return periods at specific sites for ground motions of various magnitudes. These methods are used in conjunction to help determine appropriate seismic design requirements. Similar methods are evolving for volcanic and diapiric phenomena.

The consequences of tectonic events must also be estimated. Observations of earthquake-related damage, both at surface facilities (Lew et al. 1971) and in mine tunnels (Pratt et al. 1978 and Dowding et al. 1978), provide empirical data for substantiating calculations based on the physical response properties of the structures of interest.

The consequences of such intrusive processes as salt diapirism and volcanism are estimated by studying the geometry, disruption zones, and chemical alterations associated with existing intrusions. Where conditions allow current study, the movement of faults, intrusions of material, and tectonics are evaluated also in terms of their effects on hydrologic systems and erosional processes. Impacts of faulting, erosion, and intrusion are estimated parametrically by assuming various event-scenarios and analyzing their effects on the hydrologic flow models.

The prediction of tectonic events and their potential impacts over periods of tens of thousands of years is an advancing capability. Careful selection of repository sites can reduce the likelihood of tectonically induced disruptive events to almost zero. The potential impacts of postulated events will be defined by scenario analysis in order to assess their effects on containment and isolation.

Resource Studies

The potential for exploiting mineral, energy, water, and subsurface land-use resources both now and in the future will be assessed throughout the site-selection process. Geologic, geophysical, borehole, and geochemical studies conducted during site exploration and qualification provide data for evaluating the potential for resource development. The exploration and ultimate selection of a repository are the converse of seeking an ore body or an oil field, in that investigations are conducted to locate areas with a low resource potential. If any characteristic, including thermal gradients, in the site location significantly exceeds the crustal average, its potential value to future generations needs to be carefully considered. The consequences of inadvertent human intrusion into the repository due to resource exploration at some future time must also be considered.

Status of Ongoing Exploration Programs

Preceding sections have described the factors of the natural system important in site selection, design, and construction of deep geologic repositories; the requirements that must be satisfied by a repository site; and the methods available or being developed for characterizing and assessing the natural system.

This section identifies site-specific geologic investigations conducted over the last several years. The site characterization process, described in Section 5.1.1.2, will be

conducted in four steps: national screening surveys, whose objective is to identify places that have some potential for waste isolation; regional studies, which evaluate a specific region of interest; area studies, which are conducted to characterize the areas of interest described by the regional study; and location studies, which further narrow the scope of the investigation to a site or sites.

Individual investigations are in various stages of the site-characterization process. Current investigations include 1) the Gulf Interior Region salt domes, 2) the Paradox Basin, 3) the Permian Basin, 4) the Salina Basin, 5) basalt flows at the DOE's Hanford Site, and 6) DOE's Nevada Test Site. Because of the generic nature of this Statement, details of site-specific studies are not included; for details regarding regional studies, the reader is referred to DOE's position statement to the NRC Confidence Rulemaking (DOE/NE-0007).

5.2.2 Waste Package Systems

Package components consist of the waste form, stabilizer, canister, overpack, sleeve, and backfill (Section 5.1.2).

Testing and development studies on various individual barrier components of the waste package system under expected conditions of geologic isolation have been in progress for several years. These studies have been conducted in industrial and national laboratories, as well as universities, both in this country and abroad. Most of these studies are not complete, but data and results generated during the past few years do indicate that components of the waste package system can prevent or minimize release of radionuclides to the natural system by functioning as effective chemical and physical barriers. Programs, program plans, and results are described in DOE/NE-0007 (DOE 1980).

Because of the many candidate materials for the waste package, package development programs will proceed in a logical sequence of scale and complexity. The following sequence of testing is planned:

- Initial laboratory testing using simulated waste
- Laboratory testing using real waste
- Large-scale testing in the field involving all components of the waste package.

Various aspects of the above tests have been described by the U.S. Geological Survey and DOE in the Earth Science Technical Plan (ONWI 1980), which discusses the types of data required and the sequence of laboratory, large-scale engineering, and field demonstration tests.

5.2.2.1 Waste Form

Presently, DOE has experience with spent fuel and glass as waste forms. In order to determine whether present-day spent fuel can be expected to behave satisfactorily in a geochemical environment, studies are being conducted to determine whether the release rates of waste nuclides are controlled by diffusion from UO_2 when the oxygen content of water is held to very low values (ONWI 1979). To date the information obtained from such experiments indicates that lowering the oxygen content of the water can significantly decrease the

release rate of the nuclides. Preliminary results indicate that, although some radionuclides are released more rapidly than others as a function of experimental conditions, spent fuel is a durable waste form that exhibits low release of radionuclides when subjected to ground water under normal repository conditions.

Historically, glass, particularly borosilicate glass, has been the major focus of alternate waste form work, and in 1977 it was selected as the reference material for immobilization of the Savannah River Plant high-level waste (Stone et al. 1979). Small-scale operating facilities have demonstrated practicality of the vitrification process (EPRI 1979). In addition to U.S. work, studies and pilot plants involving glass are under way in France, Germany, Belgium, and England. Recently, however, more attention has been devoted to other waste forms, and studies are being conducted to evaluate their characteristics (DOE 1979).

A number of other waste forms are being studied (ERDA 1976, DOE 1979). Prior to the decision to defer reprocessing, significant progress had been made in the development and testing of waste forms, such as glass, for wastes generated by commercial reactors. Subsequent to that decision, the emphasis of work on alternate waste forms has shifted to defense related wastes. DOE is continuing to sponsor work on alternate forms, and it is fully expected that the results and technology developed would be transferable, in large part, to the commercial waste program and existing liquid wastes (EPRI 1979).

5.2.2.2 Materials

For filler materials as stabilizers in spent fuel canisters, candidate materials include lead, glass, clay, sand, inert gases (e.g., helium) and castable solids (e.g., glass, lead and lead alloys, zinc and zinc alloys) (Jardine 1979 and Morgan 1974). Basic physical and chemical properties of candidate stabilizer materials are well known. Some of these candidate materials have been evaluated (under expected repository conditions) for use as barrier materials other than as stabilizers (e.g., as canister, overpack, and/or backfill barriers). Since the overall waste package functions are similar (e.g., corrosion resistance, nuclide sorptive properties, protection of the waste form), the same materials testing can, in many cases, be applied to several system components.

Canister, Overpack, and Sleeve. Candidate material selection for canister and overpack will be based largely on the results of corrosion tests as a function of temperature, radiation, and ground-water chemistry (e.g., pH, Eh, composition, and ionic strength) that are typical of the water in various media of interest (i.e., basalt, granite, salt, and shale). Applicable materials studies to date include consideration of general corrosion rates, pitting and crevice corrosion susceptibilities, stress corrosion cracking, effects of oxygen concentration, solution volume to solid surface area ratio, and possible effects from radiolysis products (Braithwaite 1979 and Magnani 1979). Filler material may also be used between the canister, overpack and sleeve.

Emplacement Shaft Backfill. Closure of the loaded repository will require backfilling the waste emplacement shaft; backfill materials are being tested for selective nuclide

sorption properties (for fission products and actinides), to significantly reduce radionuclide migration through the backfill barriers.^(a) The capability of the backfill materials to prevent or delay ground-water flow through the backfill is also being evaluated. Other properties of interest being evaluated (Neretnieks 1977 and Nowak 1979) are thermal conductivity, mechanical support strength, swelling, plastic flow, and forms and methods for emplacements (DOE Statement of Position to NRC (DOE/1980)).

5.2.3 Repository System

The repository system will provide for the receipt, inspection, transfer to the underground, emplacement, and containment after closure of radioactive wastes. Performance criteria stipulating the minimum acceptable behavior for an engineered system are required in evaluation of the design. Criteria for the performance of the mined repository during the operational phase have not yet been established; however, such criteria are expected to be similar to those for other nuclear packaging and storage facilities.

The surface facilities of a repository are similar to those now used in the nuclear industry. Radiation protection practices in the repository, therefore, will be similar to those used in other nuclear facilities and are not discussed here. Repository support facilities and underground workings are also similar in many ways to those common to the mining industry. Therefore, issues not uniquely related to radioactive waste repositories, such as the construction of support facilities, are not discussed here.

For the purpose of assessing the long-term containment and isolation integrity of a geologic repository, disruption phenomena which represent potential waste release mechanisms have been postulated. This analysis is discussed in detail in Section 5.5. Existing studies show no compelling environmental reasons, including public health, that should preclude disposal of waste in deep geologic repositories.

Other scenarios and variations of the scenarios presented in this Statement have been analyzed and published (Claiborne and Gera 1974). The conclusions of the published studies are in agreement with those provided above. However, this is a complex and extensive area of ongoing research which is generally being examined by scenario analysis, study of waste form release rate and radionuclide transport phenomena, and consequence analysis. Specific R&D projects in risk assessment are listed in Appendix L.

Discussion of potential adverse impacts of constructing and operating a repository will be limited to the following factors:

- Excavation and underground development
- Thermal effects
- Radiation effects
- Repository penetrations.

(a) Such materials are sometimes referred to as "getters" due to their ability to retard the movement of certain materials.

5.2.3.1 Excavation and Underground Development

The excavation of rooms and tunnels underground will induce a new stress state and displacement field in the host environment. The nature of these stresses and displacement fields depends on the cross-sectional geometry of the excavation, the layout of the tunnels and rooms, and the extraction ratio (the ratio of the volume removed to the volume remaining) (Koplick et al. 1979).

Fracturing around the perimeter of the tunnels and rooms and effect on in-situ stress states and its implications for long-term containment are two potential impacts being considered in the excavation of a repository.

Vast experience has been gained in the excavation of various kinds of underground facilities. Fracturing during drilling and blasting operations is limited by controlling such parameters as the size and type of charge, the configuration of drill holes, and the sequence of detonation. Controls of these types are used extensively in the excavation of underground facilities intended for storage purposes and for long-term operations (Svanholm et al. 1977); examples are caverns for compressed air and natural gas storage. In-situ tests are in progress to confirm their suitability for the excavation of mined geologic repositories (Hustrulid 1979). It is believed that no further technological advances are needed in this area (Guiffre et al. 1979).

5.2.3.2 Thermal Effects

The thermochemical impacts of principal interest in repository design are those that would accelerate the degradation of the waste package and the migration of radionuclide away from the package. The introduction of heat into the system will change the environment in which the waste was placed. The design of a waste package capable of withstanding the heat-altered environment is discussed in Section 5.1.2.

The introduction of heat into the natural system will induce stresses in the host rock and surrounding media (IRG 1979 and NAS 1979). These stresses will be superimposed on the existing stresses and must be considered in design to ensure structural stability of the repository. The heat generated by the emplaced waste will cause the rock mass to expand, thus inducing surface uplift. In the long term, as the heat generation rate decreases, the surface will subside. Displacement of the overlying rock mass may cause fracturing in the rock, thereby giving rise to perturbations in the hydrologic flow regime. In addition, heat may modify the thermal and mechanical properties of the rock; for example, an increase in temperature will enhance the ductility of a rock but reduce its ultimate strength.

5.2.3.3 Radiation Effects

The effects exerted in the host rock by irradiation have generally been considered to be of secondary importance. To date, most of the laboratory and theoretical studies have concentrated on the effects of radiation on salt. The information available on radiation effects on salt and on other geologic formations of interest for waste disposal has been

compiled (Jenks 1975). It is desirable, at this point, to conduct in-situ tests to determine the effects of radiation of interactions between the host rock and the waste package and to ascertain whether deleterious reactions occur due to synergism among the heat, radiation, and chemical interactions with the package (Carter 1979).

5.2.3.4 Repository Penetration

In general, the penetration of host rock by shafts and boreholes will be expected to have small environmental or safety consequences. Consideration of final sealing will require the evaluation of excavation techniques, the effect of excavation on the host rock (fracturing), and changes in rock stresses. Testing of plugging technology for shafts and bore-holes is in progress. Studies planned or under way addressing this matter are listed in Appendix L.

5.2.4 Summary

The following summarizes the present status of technology and R&D in support of improving the reliability of a mined geologic repository.

- The general criteria that have been proposed for repository site qualification have been identified in the "NWS Criteria for the Geologic Disposal of Nuclear Wastes: Site - Qualification Criteria (ONWI-33(2), 1980)."
- Studies of the natural geological system, development of the man-made waste package, and repository system analysis will all combine to lead to repository designs that utilize multiple barriers to their maximum efficiency in a repository.
- Regional geologic conditions in the U.S are well known and have been extensively described; geologic forecasting is being accomplished by extrapolating past geologic-event data into the future and weighing results against deterministic tectonic models.
- Ground water as the principal agent for transport of radionuclides to the biosphere has received extensive study and research; the principles that govern its occurrence and movement are well established. Additional studies are being conducted, using accepted techniques, to define regional and local ground-water flow systems.
- Sorption capacities of candidate rock media in contact with radionuclides are being determined in the laboratory. These data are designed to permit estimation of long-term migration of the radionuclides in repository host media.
- Continued development of the waste package is expected; studies with candidate materials for the waste package development will proceed in a logical sequence and scale of complexity.
- The repository system performance will be affected by excavation and underground development, thermal effects, radiation effects, and repository penetrations. These effects are being evaluated individually and synergistically for effects in overall repository performance.

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5.3 DESCRIPTION OF THE CONCEPTUAL GEOLOGIC REPOSITORY FACILITIES

In this section, descriptions are given of a conceptual repository for spent fuel in the once-through cycle and a repository for wastes from the reprocessing cycle. The conceptual repositories are described independent of geologic media with specific design and operational features that may be affected by geology detailed separately. Geologic media considered representative of formations available for location of a repository and that are described in this Statement are bedded salt, granite, shale, and basalt (other media may also be acceptable). The concept of siting repository facilities on a regional basis is also described in this section.

5.3.1 Once-Through Fuel Cycle Repository

Conceptually, a repository operating in the reference once-through fuel cycle is required to receive PWR and BWR spent fuel elements. The characteristics of these wastes are described in Section 4.2.

5.3.1.1 Design Bases

Waste emplacement at the conceptual repository is controlled by thermal criteria. The thermal criteria used here specify both areal thermal loadings, which control canister spacing, and canister thermal loadings, which limit the heat output of individual waste packages. The criteria were developed from an analysis of the thermal stresses that accumulate in the geologic formation and in the waste canisters. The criteria are designed to limit these stresses to values that will not compromise the integrity of the formation, the mine area or the waste canisters. Development of these criteria is discussed in Appendix K.

The design areal thermal loadings for the conceptual repositories for this Statement were limited to two-thirds of the calculated allowable thermal loadings. This was done to ensure a conservative estimate of capacity. These design basis thermal limits for spent fuel are shown in Table 5.3.1.

The criteria for granite and basalt, 320 kilowatts/hectare,^(a) indicate that 2.6 times more heat-generating waste may be stored in a hectare of granite or basalt than in a hectare of salt. This means that with equal areas a repository in granite or basalt would contain approximately 2.6 times more spent fuel than a repository in salt. This ratio is actually 2.4 for the conceptual repositories because of differences in the mining extraction ratios and room arrangements between the hard rocks and salt. Another parameter, discussed further in a later subsection, that affects the repository waste capacity is waste age.

We assume here that spent fuel may be sent to a geologic repository after five years of cooling. However, a large portion of the spent fuel will be considerably older and cooler. This is because of the large inventory that will accumulate before a repository is available and because of the time required to dispose of this inventory. For a 1990

(a) One hectare equals approximately 2.47 acres.

TABLE 5.3.1. Conceptual Repository Design Thermal Limits for Spent Fuel

Medium	kW/ha ^(a)	kW/acre ^(a)
Salt ^(b)	124	50
Granite	320	130
Shale	200	80
Basalt	320	130

(a) Area occupied by the emplacement rooms and their associated pillars only.

(b) The placement of spent fuel in salt is limited by long-term surface uplift. The degree of surface uplift is dependent upon the thermal loading averaged over the full emplacement area (corridor area as well as rooms and pillars). Two-thirds of the allowable average thermal loading for spent fuel in salt is 100 kW/ha (40 kW/acre). The thermal loading listed in this table (124 kW/ha) is the room and pillar area loading that results in 100 kW/ha average loading. Room and pillar integrity is the controlling criteria in other rock media and is dependent upon the room and pillar loadings listed in the table.

repository startup, the earliest date considered in this Statement, the average age of spent fuel available for the first repository was calculated to fall within the range of 7 to 11 years. For a later repository startup the spent fuel will initially be much older (See Section 7.3). For the conceptual repository described here we assume that the average age of the spent fuel delivered to the repository is 6.5 years old. The criteria in Table 5.3.1 were developed for 10-year-old fuel. Using those criteria for 6.5-year-old fuel provides an additional degree of conservatism since the thermal limit tends to increase for younger waste. There are also thermal limits for the individual canisters, but for the spent fuel repository concept used here, where the canisters contain only a single fuel assembly, the thermal output of the canisters is always well below the limit.

In the absence of detailed site-specific geologic data, optimization of the repository design to account for the special qualities of each medium is not possible. Instead a standardized repository design using a conventional underground layout is specified with an overall area of approximately 800 ha (2000 acre). This area provides reasonable waste capacity and is achievable from both construction and operational points of view. Actual repositories may be either larger or smaller than 800 ha depending upon specific site characteristics and more detailed operations analyses.

Repository design, construction, and operations presented here assume a homogeneous geologic formation without major flaws or discontinuities. This simplifying assumption is appropriate for use in this generic analysis; actual repositories will have site-specific design features. The design may involve preparation of a preliminary repository layout on the basis of initial site investigations. The preliminary layout would be modified as construction progresses and the formation is more fully explored.

For the conceptual repository described here, excavation of the full underground repository area is postulated to be completed during the first five years of repository

operation. During this period all wastes are emplaced retrievably to allow their timely removal should events during construction warrant this action. The retrievable period also provides an opportunity to evaluate the repository interface with emplaced wastes. Instrumentation will be installed to monitor temperature profiles in the waste and rock and to measure room and pillar stress and deformation. Results of these studies may verify repository design or indicate the need to modify waste emplacement procedures.

5.3.1.2 Facility Description

The conceptual repository consists of 1) surface facilities for waste receiving and handling and for mining and general operations support and 2) subsurface facilities for waste handling and emplacement and for mined rock removal. Surface facilities, shown in Figure 5.3.1, are similar for all repositories regardless of geology. These facilities and the mined rock storage pile constitute the visible evidence of the repository and occupy an area of about 180 ha at the salt and shale repositories and 280 ha at the granite and basalt repositories. The additional 100 ha at the granite and basalt repositories are required for larger amounts of rock that are mined from these formations to accommodate the additional waste disposal capacity resulting from higher thermal limits. Figure 5.3.2 provides an artist's concept of a geologic repository.

All surface structures in which radioactive wastes are handled are operated at less than atmospheric pressure. Ventilation flows are controlled by pressure differential from areas of low contamination potential to areas of successively higher contamination potential. Exhaust air is processed through a roughing filter and two high-efficiency particulate air (HEPA) filter banks in series prior to discharge via the 110 m mine ventilation stack.

Additional details of surface facilities at the repository are found in DOE/ET-0028.

The conceptual repositories for the once-through fuel cycle require three shafts in salt and shale and four shafts in granite and basalt to support waste handling and mining operations. These are the canistered waste (CW) shaft, the men and materials (M&M) shaft, and ventilation exhaust (VE) shaft in all the media and the mine production (MP) shaft in granite and basalt to support the larger mining effort.

The canistered waste shaft provides a means for transporting the canisters of spent fuel from the canistered waste building to the subsurface emplacement areas. The men and materials shaft is provided to handle mine and storage personnel, equipment, ventilation air and mined rock during excavation and backfilling. The ventilation exhaust shaft is divided into two compartments to provide separate exhaust for mining and for placement operations. The shaft discharges into the ventilation exhaust building.

The mine production shaft contains skip hoist equipment for removal of mined rock to the surface and supplies additional ventilation air to the mine.

The repository underground layout is a conventional room and pillar arrangement that serves the need for repository ventilation, opening stability, thermal effects, and efficient use of excavated space. Of the 800 ha underground area, actual spent fuel emplacement

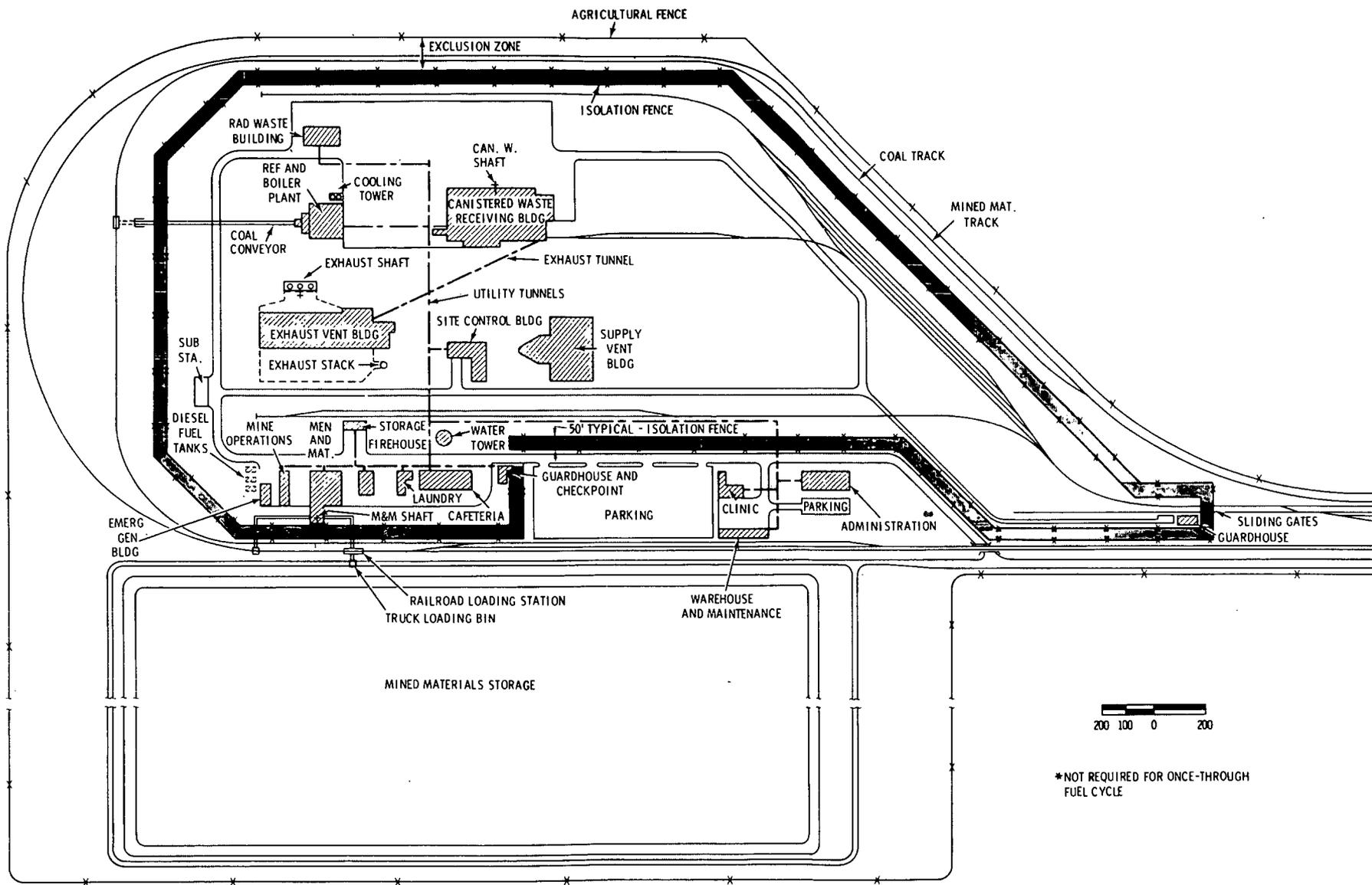
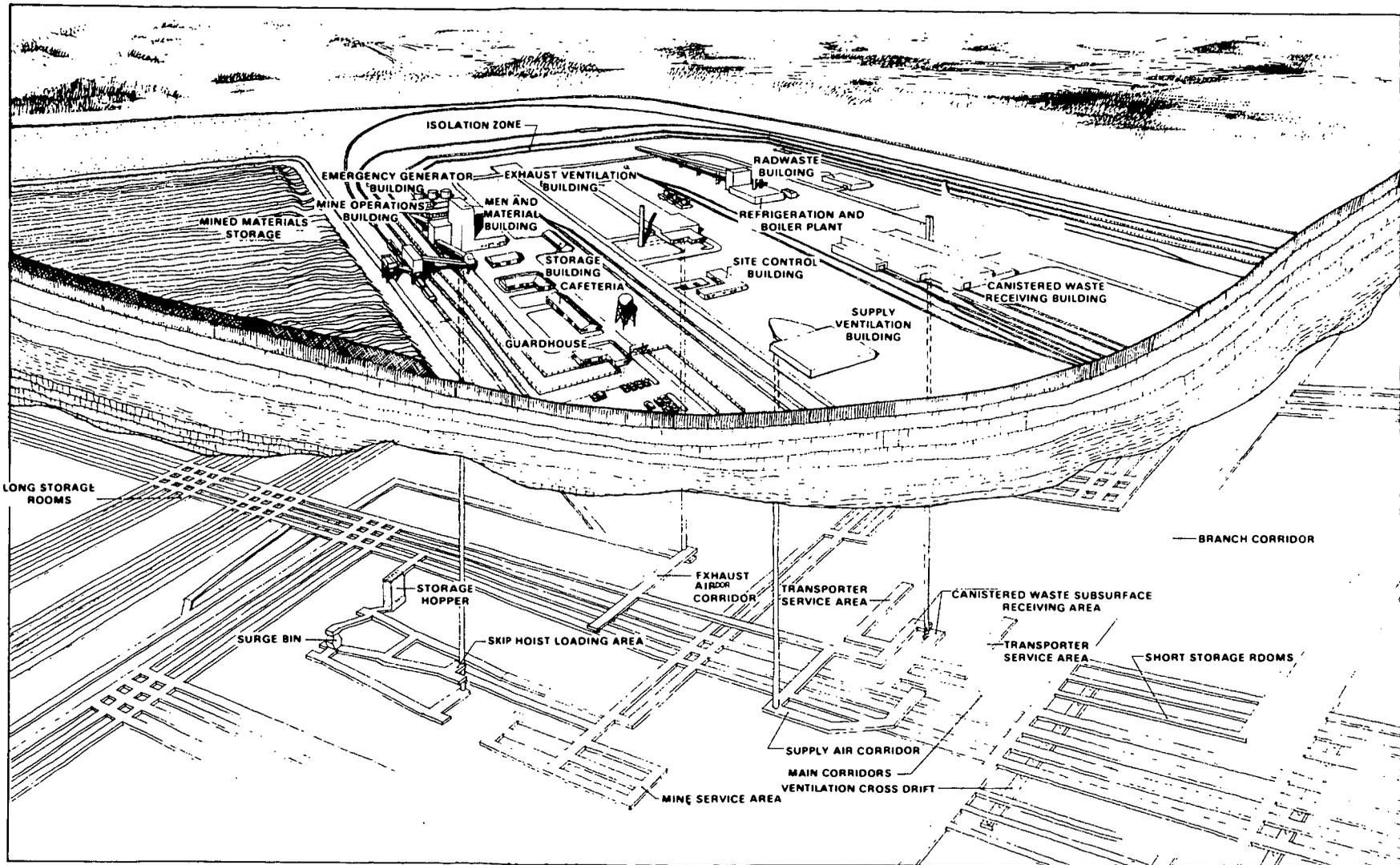


FIGURE 5.3.1. Plot Plan of a Geologic Repository



WASTE ISOLATION FACILITY

FIGURE 5.3.2. Artist's Concept of a Geologic Repository and Its Support Facilities

areas occupy 650 to 730 ha, with the remaining 80 to 160 ha occupied by shafts, general service areas, main corridors and unmined areas within the repository.

5.3.1.3 Construction

In the process of excavating repository subsurface areas, all mined rock is brought to the surface and stored onsite. The storage pile is constructed using standard earth-moving equipment. Standard dust control procedures (water sprays, etc.) are employed during construction at all repositories; salt and shale storage piles are also provided with water run-off control. When retrievable emplacement operations are complete, a portion of the rock will be returned to the mine as backfill. Present plans call for rock not used for backfill to remain piled on the surface. While in the case of a salt repository, excess salt may be disposed of by placing it in an abandoned salt mine or by selling the salt for commercial use, these options depend heavily upon the nature of specific sites. (If mined salt were to be used in commerce, the salt could be moved off site before any radioactive waste arrives onsite. Thus there would be no potential for radioactive contamination of the salt.) Quantities of rock removed and stored are described in Table 5.3.2.

TABLE 5.3.2. Mining and Rock Handling Requirements at the Reference Spent Fuel Repository

	Mined Quantity	Room Backfill	Total Backfill	Permanent Onsite Surface Storage	
	(MT x 10 ⁶)	m ³ x 10 ⁶			
Salt	30	14	17	13	6.1
Granite	77	29	38	39	15
Shale	35	15	21	14	5.5
Basalt	90	32	46	44	15

Although a repository in any of the four rock media occupies an overall area of 800 ha, larger amounts of rock are removed in constructing repositories in granite and basalt. This is due in part to larger mining extraction ratios (ratio of mined to intact volume). The increased extraction ratios are possible because of greater rock strength that allows the pillar widths to be decreased, resulting in more emplacement rooms and consequently more waste storage per given repository area.

5.3.1.4 Operations

Spent fuel packaging facilities are here assumed to be incorporated in the repository surface facilities but could be a separate facility nearby. Spent fuel elements arrive at the repository's surface facilities by rail or truck in shipping casks designed for fuel transport. These casks are lifted by crane from the rail cars or trailers to shielded transfer cells for remote removal of the spent fuel assemblies. At this point, the assemblies are examined for external contamination, signs of damage, and compatibility with other acceptance criteria. Acceptable assemblies are encased in helium-filled canisters. The helium atmosphere in the canister provides a means for canister leak testing.

Contaminated assemblies are first cleaned, then sealed in a canister; damaged assemblies are returned to their casks, transferred to the overpack cell, and encased in canisters and appropriately sized overpack canisters. The canisters are then transported to the canistered waste shaft and lowered into the repository. All spent fuel handling is done remotely.

The spent fuel canisters are received at subsurface transfer stations where shielded transporters remotely remove the canisters from the transfer stations for delivery to an emplacement room.

In addition to the thermal restrictions discussed in Section 5.3.1.1, room capacity is limited by the minimum allowable hole spacing of 1.8 m (6 ft) center to center. This is a mechanical limit that prevents weakening of the floor by holes spaced too closely together. The conceptual repositories in salt and shale emplace both PWR and BWR canisters in holes, while repositories in granite and basalt emplace PWR canisters in holes and BWR canisters in trenches. Trenches allow the relatively low heat-generating BWR canisters to be spaced more closely together (trenches are not economical for the higher heat-generating PWR canisters). The trenches run the length of emplacement rooms and contain steel racks to maintain the canisters in an upright position. They are backfilled after emplacement sleeves are installed.

Table 5.3.3 lists the contents of the conceptual spent fuel repositories in salt, granite, shale, and basalt formations at the end of emplacement.

TABLE 5.3.3. Contents of the Conceptual Spent Fuel Repositories When Full

	PWR		BWR		Total MTHM
	Canisters	MTHM	Canisters	MTHM	
Salt	68,200	31,500	104,000	19,600	51,100
Granite	162,700	75,100	246,300	46,500	121,600
Shale	86,300	39,800	131,000	24,700	64,500
Basalt	162,700	75,100	246,300	46,500	121,600

Two separate repository design concepts were also developed for the limited quantities of spent fuel, 10,000 MTHM and 48,000 MTHM, produced in the two cases (Cases 1 and 2 in Section 3.2.2) where the nuclear industry is assumed to be severely constrained. Surface facilities are reduced in size and capacity for these reduced requirements and the mined area is reduced in proportion to the quantity of spent fuel sent to disposal.

5.3.1.5 Retrievability

Actions necessary to remove emplaced wastes from a geologic repository depend on the period of repository operations during which removal takes place. Initially, wastes are emplaced in holes lined with steel sleeves and sealed with removable concrete plugs. The sleeves and plugs ensure the canisters remain accessible and minimize corrosion or other damage. During this period the wastes are considered readily retrievable in that they are removable from the repository at about the same rate and with about the same effort as for

emplacement. Beyond this initial period of operation, canisters are emplaced without sleeves and rooms are backfilled. During this later period the wastes are considered to be recoverable at considerably greater effort than emplacement.

For the conceptual repositories, readily retrievable emplacement spans the initial 5 years of operation. Repository excavation is completed during this period, and no wastes are emplaced nonretrievably until after the full extent of the repository has been explored. This provides a period for observation of waste-rock interactions when waste and local rock temperatures reach their maximum. Repository operations would also be evaluated during this period and adjustments made if necessary.

The NRC has recently proposed (Federal Register 1980)^(a) that the repository should be designed to allow retrieval of wastes for a period of 50 years after termination of waste emplacement. Whether this proposal might lead to a requirement that the wastes be readily retrievable for this period of time or recoverable has not yet been determined.

Although the specific requirements for 50-year retrievability have not yet been determined, requirements for 25-year retrievability have been estimated and the general nature of requirements for 50-year retrievability can be described. The 25-year retrievability requirements are described in Appendix K. They include use of sleeve-lined holes and concrete plugs and reduced thermal loadings for all of the spent fuel canisters. For 50-year retrievability the thermal loadings would probably have to be further reduced. An alternative approach would be to provide continuing ventilation for heat removal to reduce the rock stresses.

A particular concern for a repository in salt is closure of rooms over long retrievability periods due to accelerated "creep" deformation of the salt caused by the waste's heat. This can be compensated for, at least to some extent, by increasing ceiling heights within the repository (7.6 m height for 25-year retrievability versus 6.7 m in height for 5-year retrievability) but this may be a difficult problem for 50-year retrievability.

After repository performance has been adequately verified (after the initial 5 years of operation for these conceptual repositories, or longer if required), it was assumed that wastes would no longer be emplaced in a readily retrievable manner. For the remainder of repository operations, wastes may be emplaced in holes without steel sleeves. As the wastes are emplaced, the holes are filled with crushed rock or some specially selected backfill material. The backfill material may be an adsorptive material selected to increase the probability of long-term waste isolation. After a room is filled with waste, it is backfilled with previously excavated crushed rock or with specially selected backfill material. During this period of repository operations, the wastes are considered to be recoverable from the backfilled rooms. Recovery operations are more difficult and costly than retrieval because of the need to remove room and hole backfill. The nature of these operations increases the possibility of waste canisters being damaged before or during recovery operations but conventional techniques should be adequate. It is possible that this condition

(a) Federal Register, Vol. 4, N.94, May 13, 1980, page 31400.

might be considered adequate to meet the intent of the requirements proposed by the NRC. Additional details of retrieval and recovery operations are provided in Appendix K.

5.3.1.6 Decommissioning

As mentioned in Section 5.3.1.5, after the readily retrievable period, rooms that have been filled to capacity with spent fuel are backfilled. The technique selected for the conceptual repository is to fill the rooms with previously excavated crushed rock or with specially selected backfill material. Standard earth-moving equipment will be used to do this. This technique was selected as the most economical, and it reduces the amount of mined rock stored on the surface. With this technique, the rooms are backfilled to within 0.6 m of the ceiling with crushed rock at approximately 60% of its original density. Other backfill materials and methods of emplacement are discussed in Koplick et al. (1979).

After all rooms have been filled with spent fuel and are backfilled, the remainder of the repository underground areas are decommissioned. All corridors and underground areas are backfilled in the same manner as emplacement rooms. After this is completed, the repository shafts are decommissioned by filling to the surface and sealing. Combinations of crushed rock, clay, and concrete may be used for this purpose. Because the procedures to be used are highly site and media specific, they are not described in this generic Statement (see Koplick et al. (1979)).

Repository decommissioning is complete when the surface facilities are decontaminated, perhaps dismantled, and the repository location is monumented.

5.3.2 Reprocessing Fuel Cycle Repository

A geologic repository operating for disposal of fuel reprocessing wastes in the reprocessing fuel cycle would be required to receive high-level waste (HLW) and various remotely handled TRU (RH-TRU) and contact-handled TRU (CH-TRU) wastes. The characteristics of these wastes from reprocessing commercial fuel are described in Section 4.3. Defense program wastes could be accommodated in geologic repositories in a manner similar to that described here for these commercial fuel cycle reprocessing wastes. Characteristics and quantities of these wastes are described in Appendix I. While these latter wastes differ from those from LWR fuel reprocessing, the differences (mainly older and cooler, smaller quantities of high-atomic-number actinides and different chemical form) produce wastes with lower radiation intensities and lower heat output. Thus, repository placement criteria would be less stringent for defense wastes than those for commercial wastes and they could therefore be accommodated in the same repositories.

5.3.2.1 Design Bases

As described in Section 5.3.1.1 for the once-through fuel cycle repository, waste emplacement is subject to thermal loading criteria for a given type of waste and rock. The limits listed in Table 5.3.4 for the reprocessing fuel cycle repository are two-thirds of the calculated permissible criteria described in Appendix K.

In the case of reprocessing cycle high-level wastes there is a thermal limit for individual canisters in addition to the repository area thermal limits. These limits, which are derived from maximum temperatures, are identified in Table 5.3.5.

TABLE 5.3.4. Conceptual Repository Design Thermal Limits for Reprocessing Cycle Wastes

<u>Medium</u>	<u>kW/ha^(a)</u>	<u>kW/acre^(a)</u>
Salt ^(b)	250	100
Granite	320	130
Shale	200	80
Basalt	320	130

- (a) Area occupied by the emplacement rooms and their associated pillars only.
 (b) The placement of HLW in salt is not limited by long-term surface uplift as was the case for spent fuel in salt. Because the concentration of plutonium and its long-term heat contribution is much less in HLW, surface uplift is reduced and room and pillar integrity is the dominant concern. The integrity of rooms and pillars is dependent upon room and pillar area thermal density as listed in this table

TABLE 5.3.5. Conceptual Repository Thermal Limits for Individual HLW Waste Canisters

<u>Medium</u>	<u>Maximum kW per Canister</u>
Salt	3.2
Granite	1.7
Shale	1.2
Basalt	1.3

The conceptual repositories are designed to receive and emplace 6.5-year-old (time since reactor discharge) HLW. However, as was the case with spent fuel (Section 5.3.1.1), much of the HLW as it arrives at the repository will be older and cooler than 6.5 years. Because of this, estimates of waste emplacement for the reprocessing waste repositories are conservative because the repository could hold more waste if designed for the older and lower heat-generating rate wastes. As in the case of the spent fuel criteria, the criteria in Table 5.3.4 were developed for 10-year-old waste. Using these criteria for 6.5-year-old waste provides additional conservatism here also. However, the effect on capacity is smaller here because a substantial portion of the repository area is required for TRU wastes whose placement is not affected by the thermal criteria because they generate so little heat.

Design and construction of the conceptual fuel reprocessing waste repositories are assumed to proceed in the same manner as described for the once-through fuel cycle in Section 5.3.1.1. The overall repository area is approximately 800 ha in all cases. Construction is completed during the first five years of repository operations while all wastes are emplaced retrievably.

5.3.2.2 Facility Description

The conceptual repositories consist of surface and subsurface facilities. The surface facilities provide for waste receiving and handling, mining and general operations support. The subsurface facilities provide for waste handling and storage and mined rock removal. The surface facilities and the mined rock storage pile constitute the visible evidence of the repository and occupy an area of about 180 ha at the salt and shale repositories and 220 ha at the granite and basalt repositories. These quantities vary slightly from the spent fuel case because of different repository configurations and mining extraction ratios.

Additional details of repository surface facilities are given in DOE/ET-0028.

The conceptual geologic repositories for the fuel reprocessing wastes require the shafts described in Section 5.3.2.2 for the once-through fuel cycle repositories and an additional CH-TRU waste shaft to transfer the waste from the CH-TRU waste building to the subsurface emplacement area.

The repository underground layout is a conventional room and pillar arrangement that serves the need for repository ventilation, opening stability, thermal effects and efficient use of excavated space. Of the 800-ha total area, actual waste emplacement areas occupy 650 to 730 ha, with the remaining 80 to 160 ha occupied by shafts, general service areas, main corridors and unmined areas within the repository.

5.3.2.3 Construction

As for the once-through fuel cycle repository, all mined rock is brought to the surface during repository excavation. Mining and rock handling requirements for the conceptual repositories in the four media are compared in Table 5.3.6. The larger amounts of mined rock in granite and basalt are the result of increased mining extraction ratios in these geologies. As in the once-through cycle there is the possibility of selling the excess salt for commercial use in the case of a salt formation repository.

TABLE 5.3.6. Mining and Rock Handling Requirements at the Reference Reprocessing Waste Repository

	Mined Quantity	Room Backfill	Total Backfill	Permanent Surface Surface Storage	
	(MT x 10 ⁶)	(m ³ x 10 ⁶)			
Salt	35	15	20	15	7.1
Granite	53	17	24	29	11
Shale	30	12	17	13	5.1
Basalt	59	17	27	32	11

5.3.2.4 Operations

Canisters of HLW, and RH-TRU wastes are received and handled at the repository in a similar manner to that previously described for spent fuel in the once-through fuel cycle repository. Canisters found to be damaged or leaking are taken to an overpack cell and

sealed in an appropriately sized overpack canister. RH-TRU waste in 55-gal drums is shipped to the repository by truck, arriving in shielded Type B overpacks (see Section 4.5.3.2 for Type B overpack definition). The overpacks are lifted by crane from the truck bed to shielded transfer cells for remote removal of the drums. The drums are placed three each into steel drum-pack canisters which are sealed with a welded lid. The drum-pack is transported to the canistered waste shaft and lowered into the repository.

CH-TRU waste arrives at the repository on pallets of twelve 55-gallon drums stacked two by three by two drums high or in steel boxes measuring 1.2 x 1.8 x 1.8 m (4 x 6 x 6 ft), roughly equivalent in size to the pallet of drums. The CH-TRU is shipped by truck in special cargo carriers (see Section 4.5) loaded with three pallets or boxes of waste. The pallets and boxes are unloaded from the cargo carrier using shielded forklifts, inspected for damage and repaired if necessary, transported to the CH-TRU waste shaft and lowered into the repository.

Wastes are received at subsurface transfer stations that form integral structures with the shafts. Shielded transporters remotely remove the containers from the transfer stations for delivery to an emplacement area.

At the conceptual repositories in salt and shale formations, HLW canisters are lowered into vertical holes in the emplacement rooms in accordance with the same minimum hole spacing (1.8 m) described for spent fuel canisters in the once-through fuel cycle repositories and with an allowable thermal density calculated specifically for the HLW's characteristics. In these formations, RH-TRU waste is also emplaced in drilled holes; however the minimum hole spacing is increased to 2.3 m as a result of the larger-hole diameters necessary for the 0.76-m-diameter canisters.

The conceptual repositories in granite and basalt formations emplace HLW in vertical holes as described for the salt and shale repositories. However, RH-TRU canisters are lowered into trenches running the length of the rooms. The canisters are held upright in a single row by storage racks that allow a minimum spacing of 1 m center-to-center.

Shielded forklifts stack the CH-TRU waste pallets and boxes two high along the walls of CH-TRU waste emplacement rooms.

Table 5.3.7 lists the contents based on the example treatment processes described in Section 4.3 of conceptual repositories located in salt, granite, shale, and basalt formations at the end of operations. Because of the differences in thermal criteria the capacities of different rock media vary. For the conceptual repositories illustrated here, the relative quantities of high-level waste and TRU wastes are different on an MTHM-equivalent basis. This is because the five-year cooling hold up for the HLW resulted in a disproportionately larger quantity of TRU waste being emplaced. Subsequent repositories would fill up with more nearly equivalent amounts of HLW and TRU wastes. The capacities when equivalent quantities of HLW and TRU wastes are emplaced are also shown.

TABLE 5.3.7. Contents of the Conceptual Reprocessing Waste Repositories When Full

Waste	Salt		Granite		Shale		Basalt	
	Containers	Equivalent MTHM ^(a)						
HLW Canisters	25,800	62,200	48,000	69,000	36,000	30,500	63,000	56,000
RH-TRU Canisters	26,900	99,700	29,100	108,500	15,100	56,000	24,700	91,500
RH-TRU Drums	399,000		431,000		224,000		367,000	
CH-TRU Boxes	4,150		4,500		2,290		3,810	
CH-TRU Drums	264,000		286,000		144,000		242,000	
Capacity if Equivalent Quantities of HLW and TRU Wastes are Emplaced		71,200		78,600		41,100		73,800

(a) For the conceptual repositories the relative quantities of HLW and TRU wastes are different because the HLW is held up for a 5-year cooling period allowing a disproportionate quantity of TRU waste emplacement. The third number shows the capacity when both waste types are emplaced at the same equivalent rates.

5.3.2.5 Retrievability

These conceptual repositories are operated with the same initial period of retrievability described for the once-through fuel cycle repositories. Steel sleeves and concrete plugs are used as described for the spent fuel to protect the emplaced HLW and RH-TRU waste canisters during the retrievable period. CH-TRU waste does not require this additional protection because it is stacked compactly in the emplacement room rather than being placed into drilled holes.

5.3.2.6 Decommissioning

Reprocessing fuel cycle repositories are decommissioned in the same manner described in Section 5.3.1.6 for the once-through fuel cycle repositories.

5.3.3 Effect of Waste Age on Repository Capacity

As spent fuel or HLW ages, the intensity of emitted radiation and heat declines and the quantity of these materials that can be emplaced in a given repository area increases somewhat. Although the thermal loading criteria for a given temperature limit decreases with waste age, heat emissions from the waste decrease even faster so that the overall result is an increase in repository capacity with increasing waste age.

The thermal loading limit for 10 year old waste is smaller than the limit for younger waste (See Appendix K for details). For a fixed initial repository thermal loading, the quantity of waste is smaller and less heat will be emitted over the long term with 6.5 year old waste than with 10 year old waste. The capacities for the conceptual repositories described in the previous sections were based on 6.5-year-old spent fuel and high-level waste, conservatively employing the thermal loading criteria for 10-year-old waste. These conceptual designs were used as a conservative basis to develop environmental impacts, resource requirements and costs for individual repositories. However, in the system simulation calculations in Chapter 7, where spent fuel and HLW ages range up to more than 50 years for some of the delayed repository cases, the repository requirements are based on estimated thermal limits that vary with waste age. The limits used are two-thirds of the estimated maximum allowable loadings.

The calculated relationship between repository capacity and waste age is shown in Figure 5.3.3 for the once-through cycle and in Figure 5.3.4 for the reprocessing cycle. The capacity of a salt repository for spent fuel is indicated to be substantially less than for reprocessing wastes and increases only about 10% over the age range shown here. (Spent fuel emplacement in salt is limited by surface uplift from the long-term heat generation from the contained plutonium. This is not a problem with the other media.) Increases in capacity for the other media range from 30% for spent fuel in shale to 100% for reprocessing wastes in granite. Repository capacities for spent fuel are more than reprocessing waste capacities in granite, basalt, and shale. This is primarily because of the repository area required for TRU wastes, which ranges from 30% for 5-year-old HLW to as much as 70% in granite and basalt for 50-year-old HLW. Design optimization and/or treatments that reduce TRU waste volumes might mitigate this effect.

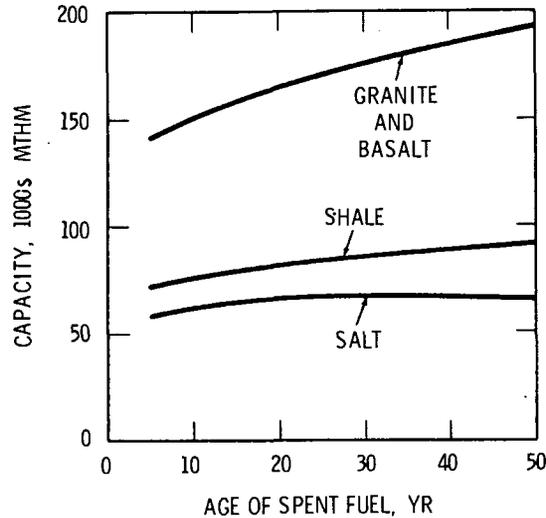


FIGURE 5.3.3. Effect of Spent Fuel Age on Once-Through Cycle Repository Capacities

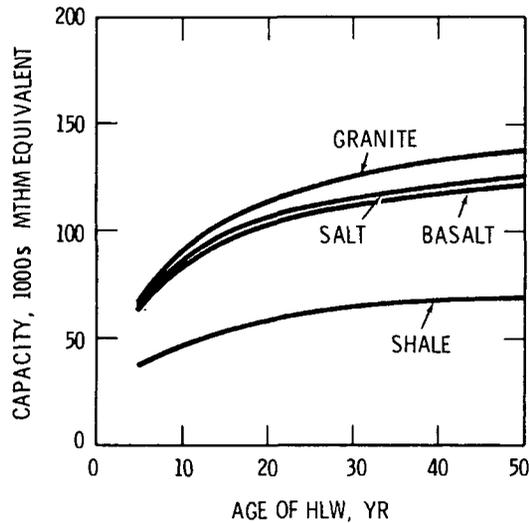


FIGURE 5.3.4. Effect of HLW Age on Reprocessing Cycle Repository Capacities

Further details regarding the basis and derivation of these repository capacities are provided in Appendix K.

5.3.4 Regional Repository Concept^(a)

To the extent permitted by availability of suitable geologic sites, two or more repositories could be located to provide disposal services on a regional basis. A regional siting concept for geologic repositories was proposed by the Interagency Review Group (IRG) on Nuclear Waste Management (IRG 1979). In its Report to the President, the IRG recommended construction of several repositories sited on a regional basis insofar as technical consid-

(a) Section 2.3 describes the present National Site Characterization and Selection Plan. Section 5.2 and Appendix B.7 discuss the technical considerations of repository site selection.

erations permit, as opposed to a single national repository. This strategy would integrate societal and political concerns as well as technical considerations.

Possible advantages of the regional concept include:

- More equitable distribution of waste management costs;
- Enhanced ability to gain public and political acceptance through cooperative participation with state and local officials and groups;
- Experience with various environments and emplacement geologic media sooner than previously planned, especially with near simultaneous development of several repositories; and
- Reduction of transportation requirements and attendant risks.

Definition of regions for nuclear waste isolation can be influenced by a number of technical, societal, and political factors. The major technical factor is the geographic distribution of acceptable geologies, but a number of other factors must also be considered.

An obvious regional division of the U.S. is one based upon individual states or combinations of states. The predominant factors that affect regional boundaries derived from the boundaries of states are the historical, social, geographical, and political factors that have existed to define the states themselves.

Regions established strictly on existing political or commercial factors could yield a wide region-to-region variation in the quantities of waste generated. Thus, there is some incentive to develop a regional structure that is based on reasonably uniform waste generation. Locations of nuclear generating capacity or electrical usage may provide an equitable basis for regional structures. Extensive electrical grid interconnections may extend the use of nuclear generated power far beyond plant locations and should be considered.

Although multiple sites themselves (except to the extent provided by different geologies) provide no guarantee against errors in disposal technology or repository design, they do help minimize the consequences of errors if the resulting failures are random and widely spaced in location and time (i.e., well after the repositories have been sealed). The potential for reduced consequences lies in the possibility of some repositories remaining unaffected, and the use of knowledge gained from the first incident to prevent subsequent incidents at other locations.

While at the present time the Department of Energy is not able to propose a specific regional siting program, regional siting is presently considered, among other factors, in the site-selection process. The Department is continuing to study the regional siting concept and should a regional siting plan be adopted, the data from the first repository could be incorporated in such a plan.

REFERENCES FOR 5.3

- Interagency Review Group. 1979. Report to the President by the Interagency Review Group on Nuclear Waste Management, TID-29442. U.S. Department of Energy, Washington, D.C.
- Koplick, C. M. et al. 1979. Information Base for Waste Repository Design, Vol. 1 and 5, NUREG/CR-0495 (7 volumes). Analytic Sciences Corp. for the U.S. Nuclear Regulatory Commission, Reading, Massachusetts.
- U.S. Department of Energy. 1979. Technology for Commercial Radioactive Waste Management, Vol. 4, DOE/ET-0028 (5 volumes). Washington, D.C.

5.4 ENVIRONMENTAL IMPACTS RELATED TO REPOSITORY CONSTRUCTION AND OPERATION

Environmental impacts related to repository construction are those estimated for construction of surface facilities and mining of the entire repository, whereas those for operation are associated with waste emplacement, backfilling and decommissioning of surface facilities. Additional details are presented in DOE/ET-0029.

5.4.1 Resource Commitments

Land use commitments for single conceptual repositories in the four geologic media are summarized in Table 5.4.1 for both spent fuel and reprocessing wastes. Other resource commitments are tabulated in Table 5.4.2 for spent fuel repositories and in Table 5.4.3 for reprocessing waste repositories. The same size (areal extent) of repository (800 ha) is postulated for each rock type; however, thermal criteria (heat loading of rock) allow spent fuel containers to be stored closer together in granite and basalt than in salt and shale, thus greater quantities of high-level waste can be stored in granite and basalt repositories for a given area than in salt and shale repositories.^(a)

TABLE 5.4.1. Land Use Commitments For Construction of 800-ha Single Geologic Repositories

<u>Land Use</u>	<u>Salt & Shale</u>	<u>Granite & Basalt</u>
Surface facilities, ha		
Spent fuel repository	180	280
Reprocessing waste repository	180	220
Access roads and railroads, ha	8	8
Mineral and surface rights, ha (fenced restricted area)	800	800
Additional land on which only subsurface activities will be restricted, ha	3,200	3,200

Land use conflicts will be highly site specific; however, most restrictions on surface use of land need not continue after repository closure. Thus, most uses of the land could resume after decommissioning of the surface facilities.

Water used during construction of a repository will range from about 1×10^5 to $5 \times 10^5 \text{ m}^3$ (depending on geologic medium) over the 7-yr construction period. As long as water can be supplied from rivers such as the R River in the midwest reference environment (Appendix F), water use will represent a small fraction (0.001) of the average river flow

(a) Note, however, that waste emplacement has not been optimized in an engineering sense for this generic Statement.

TABLE 5.4.2. Resource Commitments Necessary for Construction of a Spent Fuel Repository in Salt, Granite, Shale, and Basalt

Resource	Salt (51,000 MTHM)	Granite (122,000 MTHM)	Shale (64,000 MTHM)	Basalt (122,000 MTHM)
Water Use, m ³	240,000	710,000	360,000	610,000
Materials				
Concrete, m ³	100,000	300,000	150,000	250,000
Steel, MT	16,000	48,000	24,000	40,000
Copper, MT	220	660	330	560
Zinc, MT	55	160	80	140
Aluminum, MT	41	120	64	110
Lumber, m ³	2,300	6,900	3,000	5,900
Energy Resources				
Propane, m ³	2,200	6,400	3,200	5,400
Diesel fuel, m ³	22,000	64,000	32,000	54,000
Gasoline, m ³	16,000	47,000	21,000	40,000
Electricity				
Peak demand, kW	3,400	11,000	5,100	8,800
Total consumption, kWh	14,000,000	43,000,000	21,000,000	36,000,000
Manpower, man-yr	10,000	30,000	14,000	37,000

TABLE 5.4.3. Resource Commitments Necessary for Construction of a Fuel Reprocessing Waste Repository in Salt, Granite, Shale, and Basalt^(a)

Resource	Salt (62,000 MTHM HLW)	Granite (69,000 MTHM HLW)	Shale (30,000 MTHM HLW)	Basalt (56,000 MTHM HLW)
Water use, m ³	270,000	510,000	290,000	450,000
Materials				
Concrete, m ³	110,000	210,000	120,000	190,000
Steel, MT	18,000	33,000	19,000	30,000
Copper, MT	240	470	260	420
Zinc, MT	62	120	67	110
Aluminum, MT	46	90	50	77
Lumber, m ³	2,600	4,900	2,800	4,400
Energy resources				
Propane, m ³	2,400	4,500	2,600	4,000
Diesel fuel, m ³	24,000	45,000	26,000	40,000
Gasoline, m ³	18,000	33,000	19,000	30,000
Electricity				
Peak demand, kW	3,900	7,300	4,100	6,600
Total Consumption, kWh	16,000,000	30,000,000	17,000,000	27,000,000
Manpower, man-yr	11,000	22,000	13,000	26,000

(a) Only HLW are indicated in this and subsequent tables referring to reprocessing wastes sent to repositories. In addition to HLW, about 100,000 MTHM equivalent of TRU wastes are placed in the "first" salt repository and about 110,000, 56,000 and 92,000 MTHM equivalent in "first" repositories in other media, respectively. Subsequent repositories would undoubtedly receive a different mix of HLW and TRU wastes.

and no significant impacts are expected from its withdrawal. If a repository was to be built in an arid region, water might need to be transported to the site from areas of abundant supply.

5.4.2 Nonradiological Effluents

Nonradiological effluents from repository construction include dust and pollutants generated from machinery operation during surface facility construction and mining operations. Burning the quantities of fossil fuels listed in Tables 5.4.2 and 5.4.3 results in air pollutant emissions, but concentrations in air at the fence line are not expected to result in any air quality degradation outside applicable limits (40 CFR 50). Estimates of pollutant totals released to the atmosphere from operating equipment during construction are given in Table 5.4.4. These quantities are developed from the total quantities of fuel burned and emission factors for a given effluent (URS 1977).

TABLE 5.4.4. Quantities of Effluents Released to the Atmosphere During Construction of a Geologic Repository

Pollutant, MT	for Spent Fuel			
	Salt (51,000 MTHM)	Granite (122,000 MTHM)	Shale (64,000 MTHM)	Basalt (122,000 MTHM)
CO	7,900	23,000	10,000	20,000
Hydrocarbons	360	1,100	480	890
NO _x	1,500	4,500	2,200	3,800
SO _x	92	270	130	230
Particulates	94	270	130	230
Pollutant, MT	for Reprocessing Wastes			
	(62,000 MTHM)	(69,000 MTHM)	(30,000 MTHM)	(56,000 MTHM)
CO	8,800	16,000	9,300	15,000
Hydrocarbons	400	740	420	660
NO _x	1,700	3,100	1,800	2,800
SO _x	100	190	110	170
Particulates	100	190	110	170

Emissions from oil burning space heaters in a town of 30,000 population (about 8,000 heaters) were estimated for a 20-yr period (the approximate time surface facilities at a repository are operating) in an effort to provide some perspective for effluents released during construction of a repository. The calculated emissions were:

CO, MT	220
Hydrocarbons, MT	120
NO _x , MT	540
Particulates, MT	6,000
SO _x , MT	460

Dust from mining and rock transport within the mine is removed by filters in the mine ventilation system. However, dust generated from surface operations and rock transport to storage will result in above-ground dust. Potential dust emissions were determined using emission factors estimated by Cowherd et al. (1974). These factors were measured for rock aggregate storage piles (but not for salt) under dry and windy conditions when the dust generating potential was near maximum. Table 5.4.5 presents dust emissions for the various host rock types for both the reference environment (moist regions) and arid regions.

TABLE 5.4.5. Maximum Dust Emissions From Surface Handling of Mined Material, MT/d(a)

Climate	Spent Fuel Repository			
	Salt (51,000 MTHM)	Granite (122,000 MTHM)	Shale (64,000 MTHM)	Basalt (122,000 MTHM)
Reference	3.1	7.9	3.7	9.3
Arid	44	110	51	130
Climate	Reprocessing Waste Repository			
	(62,000 MTHM)	(69,000 MTHM)	(30,000 MTHM)	(56,000 MTHM)
Reference	3.6	5.6	3.1	6.1
Arid	49	79	44	86

(a) Assuming no control techniques are applied.

The maximum and average concentrations of dust at the repository fenceline (1.6 km from repository center) were calculated using the average annual dispersion factors (X/Q') presented for the reference environment. Table 5.4.6 presents these concentrations for the four geologic media.

The existing primary Federal air quality standard for suspended particulate matter computed as an annual geometric mean is 75 g/m^3 . Thus, for both the reference site and any proposed arid site, appropriate control techniques will be necessary to assure this limit is not exceeded during surface handling of mined material.

To give perspective to the salt concentrations at the repository fenceline, as given in Table 5.4.6, note that nearshore salt concentrations on the eastern seaboard average about $140 \text{ } \mu\text{g/m}^3$ at 0.5 km inland and about one-tenth of that 1 km inland. During persistently high onshore winds, the concentration may be on the order of $380 \text{ } \mu\text{g/m}^3$ at 0.5 km and $60 \text{ } \mu\text{g/m}^3$ at 1 km (CONF 740302 1974, pp 353-369).

Table 5.4.7 presents estimates of dust deposition rates from surface handling of mined material. Maximum deposition of dust would occur at a distance of 0.4 km from surface handling operations. At the repository fenceline (1.6 km from the handling operations) deposition is approximately a factor of 10 less. These depositions are based on the "worst case," which would consider the maximum removal rate for a year's period. Impacts of these depositions were they to occur are discussed later in the section on evaluating ecological effects of repository construction.

TABLE 5.4.6. Particulate Concentrations at Repository Fenceline, $\mu\text{g}/\text{m}^3$ (a)

Spent Fuel Repositories			Reprocessing Waste Repositories		
Repository Medium	Maximum	Average	Repository Medium	Maximum	Average
<u>Salt</u>			<u>Salt</u>		
● Reference environment	110	66	● Reference environment	130	71
● Arid environment	1400	790	● Arid environment	1600	930
<u>Granite</u>			<u>Granite</u>		
● Reference	290	170	● Reference	200	120
● Arid	3500	2100	● Arid	2400	1400
<u>Shale</u>			<u>Shale</u>		
● Reference	130	79	● Reference	110	66
● Arid	1600	930	● Arid	1400	790
<u>Basalt</u>			<u>Basalt</u>		
● Reference	330	190	● Reference	210	130
● Arid	4100	2400	● Arid	2600	1600

(a) Assuming no control techniques are applied.

TABLE 5.4.7. Dust Depositions from Surface Handling of Mined Material, $\text{gm}/\text{m}^2\text{-yr}$ (a)

	Spent Fuel Repository		Reprocessing Waste Repository	
	Distances from Handling Operations			
	0.4 km	1.6 km	0.4 km	1.6 km
<u>Salt</u>				
● Reference environment	70	8.4	90	11
● Arid environment	870	84	1100	110
<u>Granite</u>				
● Reference	180	22	140	17
● Arid	2200	220	1700	170
<u>Shale</u>				
● Reference	82	9.8	79	9.7
● Arid	1000	98	970	97
<u>Basalt</u>				
● Reference	210	25	160	19
● Arid	2600	250	1900	190

(a) Assuming no control techniques are applied.

The main concern related to surface stockpiles would be the need to protect the ground and surface waters from being contaminated with stockpile runoff, particularly in the case of salt. For repositories in salt, one plan calls for an impermeable lining of hypalon covered by 2 ft of montmorillonite-type clay to be placed over the entire stockpile area after grading and before stockpiling begins. The hypalon and clay function as a ground-water protection barrier. Construction of a trench with the same type of protection around the stockpile could collect runoff water and transport it for any required treatment. If the mine is located in an area with an arid climate, an evaporation pond may provide the required treatment. If an evaporation pond is not practical, the runoff water may be drained into a sump and pumped to a water treatment plant where dissolved salt or other solids could be removed.

Several methods for disposing of salt in excess of needs for backfilling have been investigated (D'Applonia 1976). These included disposal at sea, backfilling abandoned mines, and use in the salt trade. Salt stockpiles crust quickly and industry does not spread asphalt or chemicals on top of stockpiles to prevent loss of salt through erosion. However, covering the piles with asphalt or rock and earth may be an appropriate means of assuring dust control in the long term. Several methods appear to control or satisfactorily reduce movement of salt by wind and water. The DOE recognizes the potential for contamination of land by salt and, if a repository is located in salt, is committed to its proper control or suitable disposal.

Shale could conceivably contain amounts of soluble minerals that would be detrimental to the environment. Precipitation could leach these minerals and pollute surface and ground waters. Moreover, in a cold climate, freezing of the wet rock might result in fragmentation and liberation of particulates, resulting in particulate pollution of the streams. The shale stockpile area could be covered with a blanket of montmorillonite clay and sloped toward a collecting ditch. The surface water would then drain into a settling pond to collect silt and sands. From the pond it would be pumped to a water treatment plant where minerals in solution would be removed before release until surface facilities are decommissioned. (At present no provision is made for water treatment after the surface facilities have been decommissioned.)

Granite and basalt generally do not contain noxious soluble substances. Therefore, the stockpile area would not need special treatment and surface water would not have to be treated.

Sanitary waste will be collected in a sewer system that is connected to a local sewer trunk, if available, or given secondary treatment at the repository and disposed of in accordance with local and Federal regulations. Storm drains will be separate from the sanitary sewer system and will lead to a storm drainage pond in the general yard area.

Although dust and nonradiological pollutants generated during construction have a recognized potential for temporary adverse effects, with proper control measures, no long-term effects are expected to result.

5.4.3 Radiological Effects

The release to the atmosphere of naturally occurring radon and its decay products will increase during mining of the repositories. Estimated quantities of these radionuclides likely to be released annually to the biosphere for the various geologic media are listed in Tables 5.4.8 and 5.4.9.

TABLE 5.4.8. Annual Releases of Naturally Occurring Radionuclides to Air for Construction of Geologic Repository for Spent Fuel, Ci

Nuclide	Geologic Media			
	Salt (51,000 MTHM)	Granite (122,000 MTHM)	Shale (64,000 MTHM)	Basalt (122,000 MTHM)
^{220}Rn	9.3×10^{-4}	2.0×10^1	6.1	3.1
^{222}Rn	1.3×10^{-3}	1.9×10^1	7.0	2.7
^{210}Pb	1.1×10^{-7}	1.6×10^{-3}	5.9×10^{-4}	2.3×10^{-4}
^{212}Pb	1.4×10^{-6}	3.0×10^{-2}	9.2×10^{-3}	4.7×10^{-3}
^{214}Pb	1.3×10^{-3}	1.9×10^1	7.0	2.7
^{210}Bi	1.3×10^{-3}	1.9×10^1	7.0	2.7

TABLE 5.4.9. Annual Releases of Naturally Occurring Radionuclides to Air for Construction of Geologic Repository for Fuel Reprocessing Waste, Ci

Nuclide	Geologic Media			
	Salt (62,000 MTHM)	Granite (69,000 MTHM)	Shale (30,000 MTHM)	Basalt (56,000 MTHM)
^{220}Rn	1.1×10^{-3}	1.4×10^1	5.1	2.0
^{222}Rn	1.6×10^{-3}	1.3×10^1	6.0	1.7
^{210}Pb	1.3×10^{-7}	1.1×10^{-3}	2.5×10^{-4}	1.4×10^{-4}
^{212}Pb	1.7×10^{-6}	2.1×10^{-2}	7.7×10^{-3}	3.0×10^{-3}
^{214}Pb	1.6×10^{-3}	1.3×10^1	6.0	1.7
^{210}Bi	1.6×10^{-3}	1.3×10^1	6.0	1.7

A summary of 70-yr whole-body doses to the construction work force and to the regional population from the releases of "enhanced" quantities of naturally occurring radionuclides is given in Table 5.4.10.

The 70-yr dose from undisturbed naturally occurring radionuclides is about 7 rem/person. The 70-yr dose to the regional population is about 14,000,000 man-rem from undisturbed naturally occurring sources.

In this report, 100 to 800 health effects are postulated to result in the exposed population per million man-rem. Based on the calculated doses to the regional population, no health effects are expected to result from construction of a geologic repository for spent fuel or for reprocessing wastes.

TABLE 5.4.10. Summary of 70-Yr Whole-Body Dose Commitments from Naturally Occurring Radioactive Sources During Mining Operations at a Repository, man-rem

Repository	Spent Fuel Repositories			
	Salt	Granite	Basalt	Shale
Work force (7 yr in the repository mine)	0.18	5000	6200	1900
Population (within 80 km)	0.007	100	15	38

5.4.4 Evaluation of Ecological Impacts Related to Repositories^(a)

Construction of surface facilities at repositories will involve the removal of vegetation and displacement of birds and small mammals from the site areas. Weedy species of plants would invade cleared areas unless revegetation practices are applied. Localized dust problems would occur until vegetation cover is re-established.

Soil erosion control measures will be needed to prevent surface runoff from adding suspended solids to nearby land and surface waters. If only reasonably good practices were used, effects from construction of the surface facilities on aquatic biota should be negligible.

5.4.4.1 Ecological Effects Related to Repositories in Salt

The major ecological impact would be from fugitive dust depositions which might occur from surface handling operations of mined material. Of most concern are the estimated salt depositions at the repository fence line of 8.4 and 84 g/m²-yr for the reference and arid environment, respectively. These depositions were calculated from the case where 3.0×10^7 MT of salt was mined with 1.3×10^7 MT remaining on the surface for final disposal.

Adverse biotic effects on vegetation would depend upon many factors, including rate of uptake, short- and long-term sensitivity of species to effluent concentrations, period of exposure, the physiological condition of the vegetation during the time exposure and buildup of salt over time. Impingement upon vegetation with subsequent foliar absorption appears to be the most hazardous mode of entry. Uptake of salt solutions by foliage is a rapid and relatively efficient process (Bukocac and Wittier 1957). Crops particularly sensitive to salt effects are alfalfa, oats, clover, wheat, Indian rye grass, and ponderosa pine. These plants are seriously damaged during germination and young-leaf stage development. Ornamental vegetation types that are susceptible to salt concentrations are dogwood, red-maple, Virginia creeper and wild black cherry. Visual symptoms of toxicity are foliar necrosis, short-time dieback and "molded" growth habits. Beans are particularly sensitive showing wilting of areas on primary leaves followed by necrosis of previously wilted areas and

(a) In the following discussion of ecological impacts it is assumed that no precautions are taken. Impacts presented can be reduced to insignificant levels through application of available engineering techniques. DOE is committed to discovery and resolution of any potentially significant specific ecological effects.

chlorosis of young trifoliolate leaves. Effects on vegetation will depend on air concentration and time of exposure as well as humidity. Generally, an air concentration above $10 \mu\text{g}/\text{m}^3$ will alter distribution and growth of plants (Bernstein and Hayward 1958). Because fence-line ground level concentrations for salt dust released from surface storage and handling operations will exceed this level, a significant affect would be expected. The deposition rates are in the range of 40 to $95 \text{ gm}/\text{m}^2/\text{yr}$ for observable leaf-burn on such plants as beans. Based on the assumptions made for determining salt depositions, mitigating procedures would be needed to reduce salt dispersal at least two orders of magnitude to ensure that emission concentrations are well below levels toxic to plant life. Once contaminated, salt-affected soils will require special remedial measures and management practices to restore them to their original productivity.

Potentially, salt would be deposited as dust on the land and would also be transported by runoff to nearby surface waters. Salt concentrations on the order of 8000 parts per million (ppm) are lethal to freshwater fish under conditions of acute exposure (Jones 1964), and the recommended limit for chronic exposure is 80 ppm or 0.01 of the acute toxicity level (NAS 1972). The possibility exists for surface waters, particularly shallow, catch basin-type ponds, to receive amounts of salt sufficient to damage indigenous aquatic plants and animals. Resident species might also be replaced by more salt-tolerant forms.

In addition to effects from dust deposition, localized effects occur from leaching around the surface storage area. Fluctuations in concentrations of soil salinity would depend on precipitation, drainage, seepage, wind and rain erosion rates, and salt concentrations in water and air that come into contact with the soil. Increased salinity around the storage area would decrease or eliminate plant growth, because high salt concentrations in soil reduce the rate at which plants absorb water. This would limit the use of vegetation to increase the aesthetic qualities of the storage area and to control dust.

5.4.4.2 Ecological Effects for a Repository in Granite

A deep geologic radioactive waste repository in granite would be potentially less ecologically damaging than a salt repository and as a consequence would require fewer mitigating measures. During construction, about 8×10^7 MT of rock would be mined and 4×10^7 MT would require disposal. For convenience of operation the granite would probably be crushed in the mine before being brought to the surface, thereby reducing the airborne dust contamination at the surface.

Possible methods of disposal include removal for use in construction projects (e.g., dams, highways) or surface disposal. Neither of these alternatives pose serious ecological problems. Apart from land use associated with surface storage of the mined material, several hundred tons of airborne particulates may be released yearly. Environmental release of this material to land or surface water could be limited by establishing a vegetation cover for the stored rock, and by proper draining and ponding the surface runoff.

During construction of a granite repository, as with shale and basalt, water may enter either through downward flow from the overlying strata or through upwelling from lower

layers. The volume of water entering the repository is generally directly related to repository size and will be greatest during the last stages of construction-operation when the repository is near its maximum size. For granite the estimated inflow of water could be about 1500 m³/day (400,000 gal/day). Much of this water will be removed as water vapor by the mine ventilation system, although some of the water will probably require collection in sumps in the mine and pumping to the surface. Nonradiological water quality standards will have to be met before this effluent is released to land or surface waters. Disposal of this water will only be necessary until the repository is sealed off. However, the maximum volume of water that would likely need treatment and disposal probably will be less than 760 m³/day and is not expected to create ecological problems.

5.4.4.3 Ecological Effects for a Repository in Shale

In the case of a deep geological repository in shale about 3.5×10^7 MT of rock would be mined and 1.4×10^7 MT would require disposal. The mined material would be crushed before it is brought to the surface, a practice that will reduce the release of dust above ground. Several disposal methods may be applicable for mined shale not required for back-filling of the mine. These methods are surface storage, ocean disposal, and placement in abandoned mines. Each of these alternatives has some potential for causing ecological impact. Mine storage may contaminate ground-water supplies that may, in turn, impact ecological systems; some local but poorly defined impacts may result from ocean disposal; and surface storage may remove land from the available natural habitat and be a source of acid runoff.

Shale may contain up to 0.5% iron pyrite, which will produce sulfuric acid when exposed to oxygen and water. Runoff from storage piles, water pumped from the mine, leaching of shale if it were disposed of in abandoned mines, storage, and ocean disposal may provide sources of this acid waste to the environment. The actual quantities and acidity of this waste water have not been defined. Potential ecological impacts will probably be localized and highly site specific. Factors such as the ambient pH of the soil and receiving water, their buffering capacity and the interaction with other physical and chemical parameters will be important in controlling the affects. To afford a moderate level of protection for aquatic life, the pH of freshwater systems should be between pH 6.0 and 9.0, and there should be no change greater than 1.0 units outside the estimated seasonal maximum and minimum (Jones 1964). In marine waters, the addition of foreign material should not reduce the pH below 6.5 or raise it above 8.5, and within the normal range the pH should not vary by more than 0.5 units. Natural plants and animal communities are found on soils ranging from acid bogs to highly alkaline arid environments, and limits of appropriate release would be site specific.

As was the case with the granite repository, shaft and mine liquid effluents are expected to seep into the shale repository during construction. The estimated maximum inflow during the last stages of construction will be about 19,000 m³/day (5,000,000 gal/day). Most of this water will be collected in sumps, pumped to the surface and treated. One or more holding ponds will be used to retain the water prior to cleanup and release to

the environment. Discharge of this volume of water to the environment could require piping or ditching to reduce erosion, and could require sufficient cleanup and neutralization of acid to prevent environmental impact.

5.4.4.4 Ecological Effects for a Repository in Basalt

The expected ecological impacts from the construction and operation of a basalt geologic repository will be small and similar to that of a granite repository. Some impact will occur from noise, dust, and disturbance of surface soil. This will be mainly confined within the 81 ha (200 acre) control zone.

About 9.0×10^7 MT of basalt rock will be mined and 4.4×10^7 MT will require disposal. Suggested disposal methods include surface storage and use in large construction projects (e.g., highways). Several hundred tons of dust will be released per year unless reduced by establishing vegetation on the spoils piles. Erosion through runoff will be controlled by ditching and catch basins. Environmental release of silts from runoff will be small, because the basalt deposits under consideration for a repository are in arid regions. Except for land use considerations, the impacts of the basalt repository will be of little ecological consequence.

5.4.4.5 Ecological Impacts Related to Repositories for Reprocessing Wastes

Ecological effects of repository construction for the reprocessing wastes are expected to be similar to those of spent fuel repositories. Impacts from salt repository construction for these fuel reprocessing wastes are slightly greater than for spent fuel because about 20% more salt is mined. Impacts of granite, shale, and basalt repository construction are less than impacts of spent fuel disposal, because about 32%, 15%, and 34% less materials, respectively, are mined. Again the major ecological impact is from dust depositions that occur from surface handling operations of mined material. Of major concern is the potential for salt depositions at the salt repository fence line of 11 and 110 g/m²-yr for the reference and arid environments, respectively.

5.4.5 Nonradiological Accidents

Table 5.4.11 summarizes the number of predicted injuries (temporarily disabling) and fatalities (or permanently disabling injuries) associated with surface facility construction and underground mining operations for the various geologic media for spent fuel and fuel reprocessing waste repositories. These predictions are based on an injury rate of 13.6 temporary disabling injuries per million hours of construction (National Safety Council 1974) for the surface facilities, and an injury rate of 25 temporary disabling injuries per million man-hours for underground mining (other than coal). A fatality rate of 0.17 fatalities (or permanently disabling injuries) per million man-hours of construction (same site) for the surface facilities and 0.53 fatalities per million man-hours for underground mining (other than coal) were used.

Normalizing the construction injuries and fatalities based on standard industrial statistics to a 100,000 MTHM spent fuel repository, the injuries by rock type are about 860,

TABLE 5.4.11. Estimates of Nonradiological Disabling Injuries and Fatalities Associated with Repository Construction Based Upon Current Industrial Statistics for Similar Operations^(a)

	Spent Fuel			
	Geologic Media			
	Salt (51,000 MTHM)	Granite (122,000 MTHM)	Shale (64,000 MTHM)	Basalt (122,000 MTHM)
Surface Facility Construction				
● Disabling Injuries	70	70	70	70
● Fatalities	1	1	1	1
Underground Mining Operations				
● Disabling Injuries	370	1400	580	1700
● Fatalities	<u>8</u>	<u>30</u>	<u>12</u>	<u>37</u>
Total				
● Disabling Injuries	440	1500	650	1800
● Fatalities	9	31	13	38
	Fuel Reprocessing Waste			
	Salt (62,000 MTHM)	Granite (69,000 MTHM)	Shale (30,000 MTHM)	Basalt (59,000 MTHM)
	Salt (62,000 MTHM)	Granite (69,000 MTHM)	Shale (30,000 MTHM)	Basalt (59,000 MTHM)
Surface Facility Construction				
● Disabling Injuries	84	84	84	84
● Fatalities	1	1	1	1
Underground Mining Operations				
● Disabling Injuries	420	1000	510	1200
● Fatalities	<u>9</u>	<u>21</u>	<u>11</u>	<u>25</u>
Total				
● Disabling Injuries	500	1100	590	1300
● Fatalities	10	22	12	26

(a) Disabling injuries include only temporary disabling injuries; fatalities include permanent disabling injuries.

1200, 1000 and 1500 for salt, granite, shale and basalt, respectively; fatalities amount to about 18, 25, 20, and 31 for salt, granite, shale and basalt, respectively. These losses need to be recognized as perhaps the largest impact associated with the routine management of radioactive wastes, and DOE plans for rigorously enforced safety programs to reduce these potential losses.

5.4.6 Environmental Effects Related to Repository Operation

The operational phase of spent fuel repositories will include the receiving, handling, and placement of spent fuel elements into assigned subterranean storage areas and the subsequent backfilling of these areas when they reach capacity. Similarly, the operational phase of the repositories for reprocessing fuel cycle wastes includes the receiving and handling

of wastes, placement of waste canisters and other containers into assigned subterranean storage areas, and the subsequent backfilling of these areas when full.

5.4.6.1 Resource Commitments

Resource commitments for operation of a geologic repository for spent fuel are summarized in Table 5.4.12. Resource commitments for operation of a geologic repository for fuel reprocessing wastes are summarized in Table 5.4.13.

TABLE 5.4.12 Resource Commitments for the Operational Phase of Spent Fuel Geologic Repositories

<u>Materials</u>	<u>Salt</u> (51,000 MTHM)	<u>Granite</u> (122,000 MTHM)	<u>Shale</u> (64,000 MTHM)	<u>Basalt</u> (122,000 MTHM)
PWR canister overpacks, steel, MT	2.5×10^1	5.4×10^1	2.8×10^1	5.4×10^1
BWR canister overpacks, steel, MT	2.8×10^1	6.2×10^1	3.6×10^1	6.2×10^1
PWR retrievability sleeves (5-yr only) steel, MT	8.8×10^3	8.8×10^3	8.8×10^3	8.8×10^3
BWR retrievability sleeves (5-yr only) steel, MT	1.0×10^4	1.4×10^5	1.0×10^4	1.4×10^5
PWR concrete plugs (5-yr only), MT	7.5×10^3	7.5×10^3	7.5×10^3	7.5×10^3
BWR concrete plugs (5-yr only), MT	7.4×10^3	7.4×10^3	7.4×10^3	7.4×10^3
<u>Energy</u>				
Electricity (kWh)	1.5×10^9	3.2×10^9	1.7×10^9	3.2×10^9
Diesel fuel (m ³)	2.1×10^5	3.2×10^5	2.3×10^5	3.2×10^5
Coal (MT)	1.2×10^6	1.8×10^6	1.3×10^6	1.8×10^6
Manpower (man-years)	1.1×10^4	2.0×10^4	1.3×10^4	1.9×10^4

5.4.6.2 Nonradiological Effluents

The major nonradiological effluent from facility operation would be fugitive dust emissions from surface handling of mined materials, as was discussed under construction impacts (Section 5.4.4). Other nonradiological pollutants released to the biosphere during the repository's operational life are given in Tables 5.4.14 and 5.4.15 for the various geologic media. These pollutants include combustion products from burning diesel fuel (URS 1977) during underground mining operations and from surface burning of coal (OWI 1978).

The estimated releases of pollutants from a geologic repository as given in Table 5.4.14 would not, in any case, result in Federal Air Quality Standards being exceeded at the repository boundary. For example, the maximum concentration of particulates at the repository boundary (1.6 km from point of release, where the \bar{x}/Q is 1×10^{-6} sec/m³) was estimated to be $0.8 \mu\text{g}/\text{m}^3$ compared to the standard of $75 \mu\text{g}/\text{m}^3$.

Heat released from buried nuclear waste will increase the temperature of the geologic formation in which it is buried and may alter the physical and chemical properties of the

TABLE 5.4.13. Resource Commitments for the Operational Phase of Fuel Reprocessing Waste Geologic Repositories

Materials	Geologic Medium			
	Salt (62,000 MTHM)	Granite (69,000 MTHM)	Shale (30,000 MTHM)	Basalt (56,000 MTHM)
HLW canister overpacks, MT steel(a,b)	6.4	8.2	4.8	9.0
RH-TRU canister overpacks, MT steel	1.5×10^1	1.6×10^1	1.0×10^1	1.4×10^1
RH-TRU drum packs, MT steel	5.3×10^4	5.8×10^4	3.0×10^4	4.9×10^4
HLW retrievability sleeves, MT steel(b,c)	7.3×10^2	9.6×10^2	1.3×10^3	1.3×10^3
RH-TRU retrievability sleeves, MT steel(c)	2.9×10^4	1.9×10^5	2.9×10^4	1.6×10^5
HLW concrete plug,(c) MT	8.0×10^2	1.0×10^3	1.4×10^3	1.4×10^3
RH-TRU concrete plug, MT concrete(c)	7.2×10^4	7.2×10^4	7.2×10^4	7.2×10^4
<u>Energy</u>				
Electricity, kWh	2.1×10^9	2.6×10^9	1.4×10^9	2.3×10^9
Coal, MT	1.4×10^6	1.4×10^6	9.4×10^5	1.3×10^6
Diesel fuel, m ³	2.5×10^5	2.6×10^5	1.7×10^5	2.3×10^5
Steam, MT	1.5×10^7	1.6×10^7	1.0×10^7	1.4×10^7
Manpower, man-yr	1.9×10^4	2.4×10^4	1.3×10^4	2.1×10^4

(a) Overpack requirements are based on 0.1% of canisters received leaking or damaged.

(b) HLW canister and sleeve diameters change with time as necessary to maintain canister heat output within limits.

(c) Sleeves and plugs needed for first five years only.

TABLE 5.4.14. Total Quantities of Effluents Released to the Atmosphere During Operation of a Geologic Repository for Spent Fuel

Effluent	Geologic Medium			
	Salt	Granite	Shale	Basalt
Particulates, MT	430	670	480	670
SO _x , MT	9,700	15,000	11,000	15,000
CO, MT	2,400	3,700	2,700	3,700
Hydrocarbons, MT	870	1,400	980	1,400
NO _x , MT	15,000	24,000	17,000	24,000
Heat, MJ	3.9×10^8	9.3×10^8	4.9×10^8	9.3×10^8

formation. The heat will eventually be transferred to the atmosphere and, if the temperatures and temperature gradients have not exceeded values that would cause damage to the formation or adversely affect the containment integrity or the environment, the formation will return essentially to its initial state. The maximum surface temperature increase in any case is not expected to exceed about 0.5°C. This aspect is discussed more fully in Section 5.5 and in DOE/ET-0029.

TABLE 5.4.15 Total Quantities of Effluents Released to the Atmosphere During Operation of Geologic Repository for Reprocessing Wastes

Effluent	Geologic Medium			
	Salt	Granite	Shale	Basalt
Particulates, MT	510	540	350	480
SO _x , MT	12,000	12,000	7,800	11,000
CO, MT	2,900	3,000	2,000	2,700
Hydrocarbons, MT	1,000	1,100	710	980
NO _x , MT	17,000	19,000	12,000	17,000
Heat, MJ	7.6 x 10 ⁸	8.3 x 10 ⁸	4.3 x 10 ⁸	7.0 x 10 ⁸

5.4.6.3 Radiological Releases

Routine radiological releases from geologic repositories during normal operation will consist principally of radon emanating from exposed rock faces and radon's decay products. These releases will also occur from backfilling operations but are negligible compared to radon releases during repository construction. Occasionally, external contamination may occur on canisters as a result of some minor accident. The population dose from decontamination activities would be much less than that from operation at a spent fuel packaging and storing facility, for which the 70-yr whole-body population dose was determined to be about 1 man-rem (DOE/ET-0029).

Doses to the work force during repository operation will include contributions from receiving, handling, and placement of waste canisters into subterranean storage areas. Doses estimated to result from operations, based on expected time of operation and permissible exposure limits, are presented below for disposal of wastes for the various geologic media:

Geologic Media	70-Year Whole-Body Dose (man-rem)	
	Spent Fuel Repository	Reprocessing Waste Repository
Salt	4.3 x 10 ³	1.4 x 10 ⁵
Granite	1.1 x 10 ⁴	1.6 x 10 ⁵
Shale	5.6 x 10 ³	8.0 x 10 ⁴
Basalt	1.1 x 10 ⁴	1.3 x 10 ⁵

Radiation-related health effects using the conversion factor of 100 to 800 health effects per million man-rem (Appendix E) suggests a range of zero to 130 health effects among a workforce of about 8000. The doses tabulated suggest individual worker doses of about 1 rem per year over a 15-year repository loading period.

5.4.6.4 Ecological Impacts

The major ecological impact of repository operation would be from the handling of mined materials at the surface during repository mining and backfilling. Impacts would be caused by the airborne transfer of mined particulates to the environment near the site. These

impacts would be greatest for the repository in salt. Mitigating procedures may be necessary to control this potential threat to the environment. Impacts of fugitive dust were discussed in Section 5.4.4.

5.4.6.5 Socioeconomic Impacts

Socioeconomic impacts associated with the construction and operation of repositories are dependent largely on the number of persons who move into the locality in which the facility will be located. Because of this, the size of the local project-generated population influx was forecasted, and estimates of their needs for locally provided social services were determined. Specific economic and fiscal impacts attributable to the development of the repository cannot be treated here because they are too site dependent.

Socioeconomic impacts also depend on site characteristics (see DOE/ET-0029, Appendix C) and the assumptions used for forecasting. Site characteristics that are especially important in influencing the size of the impacts include the availability of a skilled local labor force, secondary employment, proximity to a metropolitan area, and demographic diversity (population size, degree of urbanization, etc.) of counties in the commuting region. An additional factor in the generation of impacts is the time pattern of project-associated population change. For example, a large labor force buildup followed closely by rapidly declining project employment demand could cause serious economic and social disruptions near the site and elsewhere within the commuting region.

Impacts are estimated for three reference sites, identified as Southeast, Midwest, and Southwest (see Appendix G). These areas were chosen because they differ substantially in demographic characteristics, thus providing a reasonable range of socioeconomic impacts.

The socioeconomic model employed in this analysis first forecasts a regional population in 5-yr intervals in the absence of any project activities. This population forecast serves both as a comparative baseline and as a source for a portion of the postulated future project employment. The model takes into account both primary (project related) and secondary employment effects (such as additional retail store clerks) and incorporates as separate components spouses of members of the labor force and other dependents. Projected residences of regional migrants associated with the project are distributed to counties throughout the commuting region. The model accounts for separation and retirement from project employment and replacement by new labor force members. It also accounts for the tendency of workers and their dependents to leave the region upon job separation.

In the following analysis, impacts are presented in terms of an expected level of impact. Maximum levels of impact were also calculated and appear in DOE/ET-0029. The expected impact condition is based on the most likely value of model assumptions, whereas the maximum impact condition places an extreme but credible value on the model assumption.

Table 5.4.16 presents the manpower requirements for construction and operation of a single waste repository involving spent fuel or reprocessing of wastes.

Table 5.4.17 presents estimates of the cumulative project-related in-migrants for the three reference repository sites in salt. Similar estimates were made for granite, shale,

TABLE 5.4.16 Estimated Manpower Requirements for Construction and Operation of a Single Waste Repository, by Disposal Medium and Average Annual Employment (3-yr. peak)

Medium	Spent Fuel Repository		Reprocessing Waste Repository	
	Construction	Operation	Construction	Operation
Salt	1700	870	2000	1300
Granite	4200	1100	3000	1300
Shale	2200	880	2100	1200
Basalt	5000	1100	3800	1500

TABLE 5.4.17. Forecasts of Expected Population Influx for a Geologic Repository in Salt (51,000 MTHM Waste Capacity): Number of Persons and Percent of Base Population^(a)

	Site	1980	1985	2000	2005
Spent Fuel Repository	Southeast	330 (1.9%)	540 (3.0%)	660 (3.3%)	700 (3.4%)
	Midwest	130 (0.2%)	570 (0.8%)	710 (0.9%)	740 (0.9%)
	Southwest	5,200 (10.8%)	4,200 (8.5%)	5,000 (9.2%)	5,100 (9.1%)
Reprocessing Waste Repository	Southeast	410 (2.3%)	760 (4.1%)	930 (4.6%)	980 (4.7%)
	Midwest	200 (0.4%)	860 (1.3%)	1,100 (1.3%)	1,100 (1.3%)
	Southwest	6,200 (12.4%)	5,700 (11.3%)	6,800 (12.1%)	6,900 (12.0%)

(a) The dates shown are for one possible scenario and do not attempt to reflect actual schedules. The effects of population influx are expected to be substantially the same regardless of actual startup date.

and basalt and are presented in DOE/ET-0029. The forecasted values include primary and secondary workers and associated household dependents, all of whom are in-migrants. Some of the persons who separate from the facility will stay in the site county and some will leave. Those who will stay are included in the forecasted values. Thus, not all forecasted populations are actually working on or directly associated with the project at each time period. Nevertheless, the presence of each of these persons would be caused by the existence of the project; they would probably not be present if the project did not occur. The percentages associated with each population in these tables reflect the size of the in-migrant group relative to the baseline population in the respective sites. Since these baseline populations vary by site, the relative impact of a similar in-migrant group can vary greatly.

Manpower requirements for construction of disposal facilities are lowest for a repository in salt and highest for a repository in basalt. For a spent fuel repository in salt, the total numbers of forecasted new in-migrants in the Southeast and Midwest sites under expected impact conditions are under 3% of the site county populations in the construction (1980-1984) and operation (1985-2005) phases. In-migration at this level is not likely to produce significant impacts. The effect of a repository in salt at the Southwest site is substantially different. The number of in-migrants during construction is over three times the level of primary employment demand (4200 versus 1700). Project related in-migration that exceeds 10% of the corresponding baseline population is considered to produce significant impacts. In-migration to the Southwest site exceeds this level in most cases. For a repository in granite, expected impacts at the Southeast and Midwest sites are judged to be

non-significant. Again, the Southwest site is subjected to relatively large impacts, primarily because there is a scarcity of skilled available local labor.

The translation of forecasted project-related in-migration into socioeconomic impacts is complex and imprecise. Estimates of the level of demand that will be placed on the community to provide social services to the new workers and their families were made by applying a set of factors (see DOE/ET-0029, Appendix C) to the project in-migration values. The product indicates how many units of each social service would be "expected" by the in-migrants. The severity of these impacts is primarily related to the capacity of the site county to adsorb these expected values. To contain all of the spent fuel in a 10,000 GWe-yr scenario, eight reference repositories in salt, three in granite or basalt, or six in shale were estimated to be required; thus, the impacts described would occur 8, 3, or 6 times (but in different places) depending on the medium chosen for disposal. In a similar way the impacts for construction of fuel reprocessing waste repositories would occur 6, 7 or 10 times depending on media chosen for disposal. (See Chapter 7 for numbers of repositories required in different power growth scenarios.)

The calculated level of the expected need for additional social services at the three reference sites is given for the year 2000 for spent fuel and fuel reprocessing repositories in Tables 5.4.18 through 5.4.21. Identification of social services that would likely be required indicates the potential extent of socioeconomic impacts. The ability of communities to provide services identified here, with or without financial assistance, is highly site-specific and is beyond the scope of this document. Some of the social services listed can be described as operational, such as physicians and teachers. These needs are more easily met on a temporary, less-costly basis than are those services that require major capital investment. The latter include hospital beds to the extent that hospital space is also needed, classroom space, and additional sanitary waste treatment capacity. Capital investment needs are forecast to be large, especially in the Southwest site, and to the extent that they persist over time, they will represent a serious challenge to community planners and local government. The increase in the local crime rate is only one indicator of the social disruption and a sense of a decline in social well-being experienced by community residents faced with large-scale development. This analysis does not address one site-specific but very important impact of any major construction activity; that is the impact of increased property values, increased taxes and increased commodity prices on fixed-income families.

In general, the reference Southwest site is more likely to sustain significant socioeconomic impacts compared with the other two sites, because it has a smaller available unemployed construction labor force, lacks a nearby metropolitan center, and is subject to the generation of greater secondary employment growth compared with the other sites. If a repository were to be built in an area where demographic conditions approximated that of the Southwest site, a detailed analysis of site-specific socioeconomic impacts would be needed to help prevent serious disruptions in provision of necessary social services.

TABLE 5.4.18. Selected Expected Social Service Demands Associated with Migration into the Site County Resulting from the Construction and Operation of a Geologic Repository in Salt

Selected Social Services	Year 2000					
	Spent Fuel Repository			Reprocessing Waste Repository		
	Southeast Site	Midwest Site	Southwest Site	Southeast Site	Midwest Site	Southwest Site
Health						
Physicians and dentists	1	1	5	1	2	7
Hospital and nursing care beds	3	5	20	4	8	28
Education						
Teachers	8	8	66	11	13	90
Classroom space, m ²	760	790	7,300	1,100	1,200	9,900
Sanitation						
Water treatment, m ³ /d	300	330	2,400	430	490	3,200
Liquid waste, m ³ /d	200	220	1,600	290	320	2,200
Fire and police, personnel	2	2	11	2	2	15
Recreation areas, ha	1	1	5	1	1	7
Government						
Administrative staff	1	1	4	1	1	5
Other social impacts						
Crimes (7 crime index)	25	25	240	35	40	330

TABLE 5.4.19. Selected Expected Social Service Demands Associated with Migration into the Site County Resulting from the Construction and Operation of a Geologic Repository in Granite

Selected Social Services	Year 2000					
	Spent Fuel Repository			Reprocessing Waste Repository		
	Southeast Site	Midwest Site	Southwest Site	Southeast Site	Midwest Site	Southwest Site
Health						
Physicians and dentists	1	3	9	1	2	9
Hospital and nursing care beds	4	11	35	5	11	35
Education						
Teachers	13	18	117	14	17	111
Classroom space, m ²	1,200	1,500	13,400	1,300	1,500	13,000
Sanitation						
Water treatment, m ³ /d	510	800	4,200	530	730	4,200
Liquid waste, m ³ /d	340	530	2,800	350	490	2,800
Fire and police, personnel	2	3	20	3	4	20
Recreation areas, ha	1	2	9	1	2	9
Government						
Administrative staff	1	1	7	1	1	7
Other social impacts						
Crimes (7 crime index)	40	60	430	40	60	430

TABLE 5.4.20. Selected Expected Social Service Demands Associated with Migration into the Site County Resulting from the Construction and Operation of a Geologic Repository in Shale

Selected Social Services	Year 2000					
	Spent Fuel Repository			Reprocessing Waste Repository		
	Southeast Site	Midwest Site	Southwest Site	Southeast Site	Midwest Site	Southwest Site
Health						
Physicians and dentists	1	1	6	1	2	7
Hospital and nursing care beds	3	6	22	4	8	27
Education						
Teachers	9	10	74	11	12	89
Classroom space, m ²	820	910	8,300	1,000	1,100	9,800
Sanitation						
Water treatment, m ³ /d	330	400	2,700	410	490	3,200
Liquid waste, m ³ /d	220	270	1,800	280	320	2,100
Fire and police, personnel	2	2	13	2	2	15
Recreation areas, ha	1	1	6	1	1	7
Government						
Administrative staff	1	1	4	1	1	5
Other social impacts						
Crimes (7 crime index)	30	30	280	30	40	330

TABLE 5.4.21. Selected Expected Social Service Demands Associated with Migration into the Site County Resulting from the Construction and Operation of a Geologic Repository in Basalt

Selected Social Services	Year 2000					
	Spent Fuel Repository			Reprocessing Waste Repository		
	Southeast Site	Midwest Site	Southwest Site	Southeast Site	Midwest Site	Southwest Site
Health						
Physicians and dentists	1	3	11	1	3	11
Hospital and nursing care beds	5	13	39	5	13	50
Education						
Teachers	14	21	130	15	20	132
Classroom space, m ²	1,300	1,700	15,000	1,400	1,800	14,800
Sanitation						
Water treatment, m ³ /d	550	930	4,700	600	860	4,700
Liquid waste, m ³ /d	370	620	3,100	400	570	3,200
Fire and police, personnel	3	4	22	3	4	22
Recreation areas, ha	1	2	10	1	2	10
Government						
Administrative staff	1	2	7	1	1	8
Other social impacts						
Crimes (7 crime index)	45	70	480	50	65	490

5.4.6.6 Environmental Effects Related to Postulated Radiological Accidents

Several accidents that could result in the release of radionuclides were analyzed for the spent fuel repositories. The accidents were chosen on the basis of their probability of occurrence and radiological consequences. Of accidents which might occur during the operation phase, the drop of a spent fuel canister down the repository mine shaft was most serious and its effects are presented here. Severe accidents after repository closure are treated in Section 5.5. Scenarios are provided in DOE/ET-0028.

For the accident involving a canister dropped down a repository mine shaft, radionuclides are assumed to be released to the mine atmosphere from the failed canister over a period of 1 hr. An elevator load is assumed to include four spent fuel assemblies containing 2 MTHM of spent fuel that are assumed to be ten years out of the reactor. The radioactive materials that would be released to the environment from such an accident are presented in Table 5.4.22. The releases were determined using the assumption that material released in the mine shaft passes through a roughing filter and two HEPA filters (total Decontamination Factor (DF) for particulates of 10^7) prior to release to the environment through a 110-m stack. Frequency of occurrence of the accident is postulated to be 1×10^{-5} per year.

Based on these releases, the 70-yr whole-body dose commitment to the maximum individual^(a) was calculated to be 3.5×10^{-5} rem. The 70-yr whole-body doses to the world-wide population would be 8.7 man-rem, compared with 4.5×10^{10} man-rem from naturally occurring sources.

Accidents were also postulated for the geologic repository for reprocessing wastes that might lead to release of radionuclides to the environs and are listed in Table 5.4.23. Scenarios are provided in Section 7.3.1.9 of DOE/ET-0028 and analyses of the accidents are presented in DOE/ET-0029. Non-design-basis accidents are discussed in Section 5.5.

Of the minor accidents, the contact-handled transuranic (CH-TRU) waste drum rupture accident (handling error) was considered most representative of the minor accidents. In this minor accident, a forklift operator error is assumed to result in the breach of one drum of CH-TRU waste. The accident can occur in the surface facility or in the CH-TRU waste mine shaft and has an estimated frequency of 0.15/yr. For the 0.63 MTHM equivalent contained in a single drum, a release fraction of 2.5×10^{-5} over a release time of 30 minutes was used.

Radioactive materials that would be released to the outside environment from this accident are presented in Table 5.4.24. The releases are assumed to be the same whether the accident occurs in the surface facility or the CH-TRU waste mine, since all releases would be released from a mine exhaust stack approximately 100 m high.

(a) The maximum individual is defined as a permanent resident at a location 1600 m southeast of the stack with the time-integrated atmospheric dispersion factor (E/Q) of 1.3×10^{-5} sec/m³.

TABLE 5.4.22. Radioactive Material Released to the Atmosphere from a Spent Fuel Canister Drop-Down-Mine-Shaft Accident at a Geologic Repository

Radionuclide	Ci	Radionuclide	Ci
^3H	6	^{238}Pu	4.0×10^{-6}
^{14}C	4×10^{-2}	^{239}Pu	5.8×10^{-7}
^{85}Kr	4×10^3	^{240}Pu	9.0×10^{-7}
^{90}Sr	1.0×10^{-4}	^{241}Pu	1.4×10^{-4}
^{90}Y	1.0×10^{-4}	^{241}Am	3.2×10^{-6}
^{129}I	6×10^{-3}	^{244}Cm	1.8×10^{-6}
^{137}Cs	1.5×10^{-4}		

TABLE 5.4.23. Postulated Accidents for the Geologic Repository for Reprocessing Wastes

	Accident Number	Accident
Minor	7.1	CH-TRU transuranic waste drum rupture caused by a handling error
	7.2	Minor canister failure due to rough handling
	7.3	Externally contaminated canister
	7.4	Receipt of dropped shipping cask
Moderate	7.5	Canister drop in surface facility
	7.6	Canister drop down mine shaft
	7.7	Tornado strikes salt storage piles
	7.8	CH-TRU waste drum rupture caused by mechanical damage and fire
	7.9	CH-TRU waste drum rupture caused by internal explosion

TABLE 5.4.24. Radioactive Material Released to the Atmosphere from a CH-TRU Waste Accident at the Geologic Repository for Reprocessing Wastes, Ci

Nuclide	U and Pu Recycle	Nuclide	U and Pu Recycle
^3H	6.3×10^{-6}	^{129}I	1.6×10^{-9}
^{14}C	1.6×10^{-10}	^{134}Cs	1.8×10^{-12}
^{60}Co	6.2×10^{-13}	^{137}Cs	1.4×10^{-12}
^{90}Sr	9.2×10^{-13}	^{238}Pu	8.2×10^{-12}
^{95}Nb	1.1×10^{-11}	^{239}Pu	5.4×10^{-13}
^{106}Ru	2.8×10^{-10}	^{240}Pu	1.1×10^{-12}
		^{241}Pu	2.7×10^{-10}

Based on the CH-TRU releases listed in Table 5.4.24, the 70-yr dose commitment to the maximum individual was calculated to be 1.0×10^{-12} rem, which is a number so small as to be effectively zero. For the same period, the maximum individual would receive about 7.0 rem from naturally occurring sources.

The 70-yr worldwide population dose from ^3H and ^{14}C calculated for this case is approximately 3.9×10^{-18} man-rem, which is effectively zero when compared with 4.5×10^{10} man-rem received from naturally occurring sources.

Calculations of the effect of a drop of a fuel reprocessing waste canister down the mine shaft indicated that this would be categorized as a moderate accident in terms of release outside the repository. Some of the canistered waste is assumed to be released to the mine atmosphere from four failed canisters in a time period of 1 hour. Canistered waste will be one of three forms:

- Solidified High-Level Wastes:
 - Glass (175 kg/MTHM)--13 kg of particles less than 10 μm in diameter will be released to mine filters. Postulated frequency of occurrence is $7 \times 10^{-7}/\text{yr}$.
 - Calcine (52.5 kg/MTHM)--31 kg of particles less than 10 μm will be released to mine filters. Frequency of occurrence is $7 \times 10^{-7}/\text{yr}$.
- RH-TRU Wastes--1.3 kg of Zircaloy fines less than 10 μm in diameter will reach the mine filters. The postulated frequency of occurrence is $2 \times 10^{-6}/\text{yr}$.

The radioactive materials that would be released to the outside environment for the various waste forms are presented in Tables 5.4.25. These releases were calculated assuming that material released in the mine shaft passes through a roughing filter and two HEPA filters (DF of 10^7) prior to escaping to the environment through a 110-m stack.

Doses to the maximum individual from these accidents are given in Table 5.4.26. The doses in Table 5.4.26 are insignificant in terms of the radiation dose of 7 rem the individual would have received from naturally occurring sources over the same time period.

TABLE 5.4.25. Radionuclide Releases for a Waste Canister Dropped Down a Mine Shaft at a Repository for Reprocessing Wastes, Ci

Nuclide	Glass	Calcine	Nuclide	RH-TRU
^{90}Y	3.9×10^{-4}	3.2×10^{-3}	^3H	2.5×10^{-1}
^{90}Sr	3.9×10^{-4}	3.2×10^{-3}	^{14}C	4.4×10^{-4}
^{106}Ru	4.4×10^{-5}	3.4×10^{-4}	^{60}Co	1.6×10^{-6}
$^{125\text{m}}\text{Te}$	4.8×10^{-6}	4.0×10^{-5}	^{63}Ni	1.6×10^{-7}
^{134}Cs	8.0×10^{-5}	1.3×10^{-3}	^{90}Sr	1.2×10^{-8}
^{137}Cs	6.0×10^{-4}	4.8×10^{-3}	^{54}Mn	8.1×10^{-8}
^{144}Ce	2.0×10^{-5}	1.6×10^{-4}	^{95}Nb	8.2×10^{-8}
^{154}Eu	3.6×10^{-5}	2.8×10^{-4}	^{137}Cs	1.9×10^{-8}
^{238}Pu	5.6×10^{-7}	4.4×10^{-6}	^{144}Ce	4.8×10^{-8}
^{239}Pu	1.3×10^{-8}	1.0×10^{-7}	^{238}Pu	1.1×10^{-9}
^{240}Pu	5.2×10^{-8}	4.0×10^{-7}	^{239}Pu	7.2×10^{-11}
^{241}Pu	6.4×10^{-6}	4.0×10^{-5}	^{240}Pu	1.5×10^{-10}
^{241}Am	5.2×10^{-6}	4.0×10^{-5}	^{241}Pu	3.6×10^{-8}
^{244}Cm	4.4×10^{-5}	3.5×10^{-4}	^{241}Am	1.4×10^{-10}
			^{242}Cm	2.0×10^{-9}
			^{244}Cm	1.4×10^{-9}

TABLE 5.4.26. 70-Yr Whole-Body Dose Commitments to Maximum Individual from Drop of Waste Canisters into a Geologic Repository

Waste	70-Yr Dose Commitment, rem
High-Level	
Calcine	1.2×10^{-4}
Glass	1.4×10^{-5}
RH-TRU	1.7×10^{-7}

In summary, radiological aspects of repository construction and routine operation including reasonably foreseeable accidents while filling and decommissioning the repository do not constitute a significant impact on public health and safety.

5.4.6.7 Radiological Impacts of Operating Accidents on the Work Force

In the case of reprocessing waste, the calculated first-year total-body dose to a member of the repository work force near the point of impact of four canisters of high-level waste dropped down a mine shaft would be 26,000 rem for waste in glass, about 210,000 rem for the waste in calcine form, and about 7,600 rem for the spent fuel case; all fatal doses.^(a) The exposure rate in the corridor due to contamination of surfaces would be approximately 20 R per hour from the waste (about 5 R per hour in the case of spent fuel). Such exposure rates would make decontaminating the corridor impossible by ordinary means; some sort of remote operation similar to that of dismantling a reactor core would be needed. However, design changes to the transfer stations in the repository and the use of two stages of HEPA filtration between the shaft and other portions of the mined repository would probably lower the occupational doses to repository workers to within acceptable ranges. These changes would limit the area contaminated to the transfer station and possibly the canistered waste (CW) shaft; although air flow should preclude significant contamination in the CW shaft. Limiting the contaminated area should also decrease the time required for decontamination and resumption of repository loading.

5.4.6.8 Other Environmental Impacts

An artist's rendering, based on engineering data, of the above-ground facilities associated with a geologic repository was shown in Figure 5.3.1. With the exception of the mine spoils piles, these facilities would not be expected to be any more of a detraction than any other mining or industrial facility of comparable size. Although the exclusion boundary could be viewed as a detraction in itself, the exclusion area will likely limit the visual impacts of the above-ground repository facilities.

(a) The source terms used in these calculations are believed to be unrealistically pessimistic but additional engineering analysis is necessary before the source terms can be reduced with confidence.

The spoils piles could have an adverse visual impact. If left onsite, these spoils, if piled 3 meters high (about 10 feet), would occupy about 2 to 5 km² (~1 to 2 square miles). This amount of material is equivalent to 13 to 44 million tons of rock, depending on repository host rock, and might be used in the construction of markers for the repository.

In the case of repositories in salt, little noise other than that from traffic would be expected in conjunction with repository construction. In the case of shale repositories construction would probably be performed with occasional blasting when encountering tightly bound hard portions of the rock; otherwise, as in the case of salt, little noise would be discernible at the surface. In the case of basalt and granite, almost all rock removal will require blasting and, as a consequence, considerable blast noise or, more likely, ground rumble would result. The degree of annoyance produced would depend in large part on the proximity of populated areas to the repository.

There were no identifiable sources of odor unique to the construction and operation of a geologic repository for radioactive waste. Increased air pollution from construction and commuter vehicles is expected; however, this is not expected to be experienced as odor. Stacks at the coal-fired support plants will be designed to mitigate noxious odors and ash from coal burning.

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5.5 LONG-TERM ENVIRONMENTAL CONSIDERATIONS OF GEOLOGIC DISPOSAL OF RADIOACTIVE WASTES^(a)

The objective of disposal of radioactive wastes in deep geologic repositories is to provide reasonable assurance^(b) that the radionuclides contained in these wastes in biologically significant concentrations will be permanently isolated from the human environment. The following presentation examines the likelihood and consequences of events that could compromise this objective over the millenia following repository closure.

No significant long-term physical impacts are expected to result from having placed the heat-emitting radioactive wastes in geologic repositories as described previously in this Statement whether located in salt, granite, shale or basalt formations. Although heat from decaying radionuclides will ultimately reach the surface of the earth via conduction through overlying rock, temperature rises at the surface were estimated to be less than 0.5°C in all cases. Such a temperature rise is insignificant. Heat flowing into and through the rock surrounding repositories will cause expansion of the rock and would result in some uplift at the surface. The largest uplifts (over several centuries) are expected to be on the order of 0.3 to 0.6 m (1 to 2 ft) in shale, and 1.2 to 1.5 m (4 to 5 ft) in salt at the center of the 800 ha (2000 acre) repository area.

Subsidence of the formation containing a waste repository following closure or collapse of the void spaces that remained after the mine has been backfilled (backfilled to 60% of volume) might occur at repositories in salt and shale. Uplift and subsidence are expected to occur over very long time periods, and as a consequence no impacts associated with earth movement are expected to result. For repositories located in granite and basalt, subsidence or uplift is believed unlikely.

Nuclear waste repositories will be sited, loaded, and sealed with every expectation that long-term radiological impacts will be nonexistent. There are, however, a few highly improbable events that can be postulated to take place singularly (or in combination with smaller probability events) that might result in radioactive wastes reaching the biosphere. Three kinds of events leading to release of some of the repository contents were postulated:

- direct release of contents to the atmosphere: Such release could follow volcanic activity, impact of a large meteorite or large nuclear weapon, or, on a much longer time scale, denuding of the earth to the depth of the repository by erosion or glaciation. Releases and consequences of these events are believed to be adequately represented by those of a meteorite strike; however, the probability of occurrence could be substantially different.

(a) "Long-term" as used here means hundreds to tens-of-thousands of years after the repository has been closed.

(b) "Reasonable assurance" is admittedly a subjective expression. While DOE believes that shallow land burial for spent fuel, HLW, remotely handled TRU or fuel reprocessing wastes would not give such reasonable assurance, DOE believes that at some depth isolation is reasonably assured. Depths on the order of hundreds of meters are believed to meet this requirement.

- release via water: Water might enter a repository as a result of flooding or seepage following the breach of overlying rock by such mechanisms as fracturing by faulting, nearby impact of meteorite or nuclear weapon, thermal stresses caused by decay heat from the radioactive waste, mechanical stresses resulting from adjustment of repository rock following excavations, or failure of shaft and/or bore hole seals. Plausible events can be postulated whereby water enters even a well-sited repository; far less plausible are events that would bring the potentially contaminated water back to the surface or to aquifers reasonably penetrable by wells.
- release via man-made intrusions: These might include exploratory drilling, solution mining of salt or phosphates; or cavern construction for storage of oil, industrial wastes, compressed air, etc.

Several of these events were chosen to provide a basis for estimating the risk of waste disposal to society. Events representative of the above categories are:

- meteorite impact penetrating to the waste bearing stratum^(a)
- fracturing through rock overlying the repository by faulting followed by stream flooding or slow groundwater infusion
- exploratory drilling through a waste canister
- solution mining for salt content, in the case of a repository in salt.

The event analyses that follow are based on the concept of "what if" they occur. In cases where probabilities could be assigned, they were used to provide an estimate of societal risk from the disposal of radioactive waste in deep geologic repositories. Following each accident discussion, a description of any action that may be taken to mitigate the consequences of the accident is presented.

Modeling methods used to estimate the consequences of the accidents are described in the appendices: Appendix D, Models Used in the Dose Calculations; and Appendix E, Radiologically Related Health Effects. Methods not described in the appendices are referenced in the text.

The radiological release consequences of the meteorite and faulting and flooding (ground-water transport) accidents are based on the assumption that breaches occur in the repository host rock itself and consequently, differing properties of the different host rocks do not enter into the calculation of the consequences. Therefore, the differences in consequences in terms of repository media are inventory related; results differ only because of the different amounts of waste disposed of in each repository. The amounts in the repositories were developed on the basis of waste emplacement in 800 ha per repository and are as follows:

(a) Representative in the sense of release and consequence but not necessarily in the sense of probability of occurrence.

	Spent Fuel	Fuel Reprocessing Waste	
		HLW	RH & CH-TRU
Salt	51,000 MTHM ^(a)	62,000 MTHM	100,000 MTHM
Granite	122,000	69,000	108,000
Shale	64,000	30,000	56,000
Basalt	122,000	56,000	92,000

If the amount of disposed waste, rather than the size of the repository, were held constant, the radiological consequences would be the same for each geologic medium. In other words, once the radionuclides are outside the repository proper, their movement away from the repository is governed by the same set of assumptions regardless of repository media. (This limitation of the analysis would be improved upon in site-specific analyses when site specific data or sorptive properties of adjacent rock become available.)

In the case of faulting and flooding with stream transport the assumption was made that the same amount of waste was removed by water regardless of repository medium. Repository medium affected consequences only in salt repositories; the presence of salt along with the wastes would likely preclude use of the emergent stream as a source of drinking water or food. Thus, except for the case of salt entering the biosphere with the waste radionuclides, no analysis was made of the waste repository medium's influence in the consequences of the postulated long-term events.

In the case of human intrusion by drilling, the same amount of waste was assumed to be brought to the surface regardless of repository media.

5.5.1 Repository Breach by Meteorite

Breach of a repository would be possible by a meteorite estimated to be about 25 m in diameter striking a point on the surface above the center of the repository at a speed of about 20 km/sec on impact. If the meteorite's density is 8 g/cm³ (which is representative of iron or nickel-iron meteorites), the mass of the meteorite at contact would be about 6.5 x 10⁴ MT and would have an energy equivalent to about 3 megatons of TNT. This meteorite would produce a crater roughly 2 km in diameter at the surface and 600 m deep at its deepest point. No clear evidence is available to suggest that meteorites of this size have created craters this deep over the age of the earth. On the other hand, the presence of astroblemes suggests that the earth has been hit by very large extraterrestrial bodies (Claiborne and Gera, 1978).

Temperatures at the impact point of the meteorite strike would reach millions of degrees, and most of the meteorite plus some of the surrounding rock would be vaporized. Some of the rock material would be pulverized and ejected into the air as the crater formed. Most of the ejected material would fall back into the crater and its immediate vicinity.

(a) Metric tons of heavy metal in the case of spent fuel or spent fuel equivalent in the case of reprocessing wastes.

If the meteorite had an energy equivalent of about 3 megatons of TNT, the overall effects would be somewhat like those from a nuclear weapon but without the prompt radiation effect.^(a) Thus, a shock wave as well as thermal effects could be expected. If a 3-megaton nuclear weapon were detonated, any individual residing within 4 km from the point of impact would be killed or would suffer at least second-degree burns and other injuries from the blast, falling buildings, and flying debris, etc.

Radioactive material suspended by a meteorite impact would be dispersed by two modes, developed on the basis of nuclear cratering test results. A typical cloud formation consists of a central column rising about a doughnut-shaped base surge, which rolls outward from the crater. One-half of the suspended material is dispersed in the central column and one-half is dispersed in the base cloud. For the reference midwest site, the material in the central cloud is also dispersed evenly across the eastern half of the United States and then moved around the world at high altitude. Compared to the base cloud, it does not contribute significantly to local (radius of 80 km) fallout. Because of large overpressures in air produced on impact of the meteorite, local low-altitude winds are assumed to have no effect on dispersion of material.

If the meteorite impact penetrated to a depth of 600 m, the impact is arbitrarily assumed to result in dispersion of about 1% of the repository inventory. The amounts of various radionuclides ejected depend on the length of time between repository closure and meteorite impact. This event was examined for a meteorite strike at the assumed time of repository closure (therefore maximum waste disposal inventory) and for 1000, 100,000 or 1,000,000 years thereafter. Assumptions about dispersion of radioactive material after meteorite impact are summarized below.

Ten percent of the particulate radioactive material dispersed is assumed to be of respirable size (^3H , ^{14}C , ^{85}Kr , and ^{129}I are assumed to be released as gases and all other radionuclides are assumed to be in particulate form). The remaining 90% of the particulate material falls back immediately into or near the crater and does not contribute to the regional population dose. For calculation of the dose to the regional population, the amount dispersed is also reduced by an additional one-half to account for the distribution of material between central and base clouds.

First-year and 70-year cumulative doses to the whole-body for various times of repository breach and for repositories in various media are presented in Tables 5.5.1 and 5.5.2. Doses to individual organs, a breakdown of dose by pathway, and tabulations of the radionuclides of importance in the repository are given in DOE/ET-0029. Calculated doses are directly proportional to the fraction of inventory released; thus, if it were postulated that 10% rather than 1% of the inventory was dispersed, the reported dose would be 10 times higher.

(a) There does not appear to be a direct equivalency between the energy of the meteorite and the nuclear weapon. Claiborne and Gera (1978) conclude that the largest presently deployed missile capable of carrying a 25-megaton bomb would form a 270-m crater; if a 50-megaton bomb were deployed a crater up to 500 m may be formed. Other calculations made for this Statement based on the work of Glasstone (1964) suggest that a bomb on the order of 130-megatons (air blast) would be required to produce a crater 2 km in diameter and 600 m deep.

TABLE 5.5.1. First-Year Whole-Body Dose^(a) to Maximum Individual--
Repository Breach by Meteorite Strike, rem

<u>Time of Event</u>	<u>Salt</u>	<u>Granite</u>	<u>Shale</u>	<u>Basalt</u>
Year of closure				
Spent Fuel	8.3×10^3	2.2×10^4	1.1×10^4	2.2×10^4
Reprocessing Wastes	1.1×10^4	9.2×10^3	6.5×10^3	1.1×10^4
Closure + 1000 years				
Spent Fuel	6.0	1.6×10^1	8.1	1.6×10^1
Reprocessing Wastes	6.2	5.3	3.8	6.2
Closure + 100,000 Years				
Spent Fuel	4.4	1.2×10^1	5.9	1.2×10^1
Reprocessing Wastes	1.1	9.2×10^{-1}	6.6×10^1	1.1
Closure + 1,000,000 Years				
Spent Fuel	2.5	6.6	3.3	6.6
Reprocessing Wastes	7.7×10^{-1}	6.5×10^{-1}	4.7×10^{-2}	7.7×10^{-1}

(a) Doses displayed in Tables 5.5.1 through 5.5.5 reflect relative differences in host rock media only to the extent that different amounts of waste are involved on a per-area basis.

TABLE 5.5.2. 70-Year Whole-Body Dose Commitment to Maximum Individual--
Repository Breach by Meteorite Strike, rem

<u>Time of Event</u>	<u>Salt</u>	<u>Granite</u>	<u>Shale</u>	<u>Basalt</u>
Year of closure				
Spent Fuel	3.9×10^6	1.0×10^7	5.1×10^6	1.0×10^7
Reprocessing Wastes	4.7×10^6	4.0×10^6	2.9×10^6	4.7×10^6
Closure + 1000 Years				
Spent Fuel	3.6×10^2	9.5×10^2	4.7×10^2	9.5×10^2
Reprocessing Wastes	3.6×10^2	4.3×10^2	2.2×10^2	3.6×10^2
Closure + 100,000 Years				
Spent Fuel	3.3×10^2	8.9×10^2	4.4×10^2	8.9×10^2
Reprocessing Wastes	3.0×10^1	2.5×10^1	1.8×10^1	3.0×10^1
Closure + 1,000,000 Years				
Spent Fuel	1.7×10^2	4.5×10^2	2.2×10^2	4.5×10^2
Reprocessing Wastes	9.4	7.9	5.8	9.4

The maximum individual, who is 4 km from point of impact, would not survive the initial blast of the meteorite. Regardless, doses in the first year following a release of wastes by a meteorite in the year of closure would amount to 8,000 to 22,000 rem to the whole-body, either of which as an acute dose would prove fatal.

An estimate was made of the number of persons in the reference environment surrounding the repository who could be expected to receive at least 500 rem in the first year following meteorite impact. This was done by calculating the ratio of the atmospheric dispersion coefficients at various points of the compass and the distance from the point of contact. The number of persons so exposed amounted to about 30,000 for the midwest site. If this dose is received in a short time, it would prove fatal to about half of these individuals; thus about 15,000 early radiation-related fatalities would be expected.

Doses to the maximum individual for a breach by meteorite 1000 years after closure range from about 1/3 to 3 times the currently applicable occupational limit and in terms of accidental exposure are not particularly noteworthy. Dose to the maximum individual as a function of time of repository breach decreases slowly after the first thousand years. For a breach at one million years, the dose would vary from about 1% to 100% of applicable occupational dose limits.

Doses to the regional population (2 million persons within 80 km) were calculated and are presented in Table 5.5.3.

TABLE 5.5.3. 70-Year Whole Body-Dose Commitment to the Regional Population--Repository Breach by Meteorite, man-rem

<u>Time of Event</u>	<u>Salt</u>	<u>Granite</u>	<u>Shale</u>	<u>Basalt</u>
Year of closure				
Spent Fuel	6.9×10^7	1.8×10^8	9.1×10^7	1.8×10^8
Reprocessing Wastes	6.2×10^7	5.3×10^7	3.8×10^7	6.2×10^7
Closure + 1000 Years				
Spent Fuel	1.6×10^7	4.2×10^7	2.1×10^7	4.2×10^7
Reprocessing Wastes	6.2×10^6	5.3×10^6	3.8×10^6	6.2×10^6
Closure + 100,000 Years				
Spent Fuel	2.8×10^5	7.4×10^5	3.7×10^5	7.4×10^5
Reprocessing Wastes	7.8×10^4	6.6×10^4	4.8×10^4	7.8×10^5
Closure + 100,000,000 Years				
Spent Fuel	9.4×10^4	2.5×10^5	1.2×10^5	2.5×10^5
Reprocessing Wastes	8.5×10^4	7.0×10^4	5.1×10^4	8.5×10^4

The population dose from a meteorite breach of a single repository in the year of closure would range from 3.8×10^7 to 1.8×10^8 man-rem.^(a) About 3.8×10^3 to 1.4×10^5 health effects^(b) might be expected from this event. For a breach in the year of closure, the dose to the regional population is about 1 to 10 times the dose received from naturally occurring sources.

As shown in Table 5.5.4, the dose for the second and subsequent generations (70 years per generation) of residents in the regional population is substantially smaller than that for the first generation. The range of doses for the second generation (from 1.1×10^3 to 2.8×10^3 man-rem) may be compared to the dose from naturally occurring sources over the same 70-yr period of 1.4×10^7 man-rem.

TABLE 5.5.4. 70-Year Cumulative Whole-Body Dose to First Five Generations^(a) of Regional Population--Repository Breach by Meteorite, man-rem

<u>Spent Fuel Repository</u>	<u>Salt</u>	<u>Granite</u>	<u>Shale</u>	<u>Basalt</u>
Generation				
1	6.9×10^7	1.8×10^8	9.1×10^7	1.8×10^8
2	1.1×10^3	2.8×10^3	1.4×10^3	2.8×10^3
3	2.1×10^2	5.5×10^2	2.7×10^2	5.5×10^2
4	6.3×10^1		- not calculated	
5	1.3×10^1		- not calculated	
<u>Reprocessing Waste Repository</u>				
Generation				
1	6.0×10^7	5.1×10^7	3.6×10^7	6.0×10^7
2	1.2×10^3	1.1×10^3	7.5×10^2	1.2×10^3
3	2.4×10^2	2.1×10^2	1.5×10^2	2.4×10^2
4	5.5×10^1		- not calculated	
5	1.2×10^1		- not calculated	

(a) A generation is taken here to mean 70 years. At the end of that time the population is replaced by an identical population that lives for 70 years.

Within the reference environment (midwest), 150 persons reside within 3.2 km of the repository center, the point of meteor impact. All of these people are presumed to be killed by the blast and thermal effects. A similar meteorite impacting in the metropolitan area of city G in the reference environment (50 to 80 km away) would result in about 25,000 immediate fatalities within a 3.2 km radius. No thought is apparently given by the public to the potential for societal loss from meteorites striking urban areas. Similarly little concern should be had for meteorites striking a waste repository, particularly since calculated consequences are somewhat less for the meteorite case.

- (a) Normalizing the 70-yr whole-body dose commitment from breach of a repository by meteorite to the electrical energy produced yields 5.5×10^4 man-rem/GWe-yr for the once-through fuel cycle and 3.6×10^4 man-rem/GWe-yr for the reprocessing cycle.
- (b) Using the range of 100 to 800 health effects per million man-rem conversion factor between dose and effect. See Appendix E for details.

Doses to the population of the eastern half of the United States were also calculated and are presented in Table 5.5.5. An assumption is that the prevailing winds in the upper atmosphere will move the radionuclides released during the accident in an eastward direction, which will expose about 160 million persons east of the midwest reference site. The 2 million persons in the reference population are excluded from this calculation. See DOE/ET 0029, Sec. 4.4.3, for additional assumptions used in these calculations. The largest tabulated whole-body dose to the eastern U.S. population of 1.5×10^8 man-rem from meteorite breach of spent fuel repository in the year of closure may be compared with the 1.1×10^9 man-rem this population would receive from naturally occurring radiation sources over the same time period.

TABLE 5.5.5. 70-Year Whole-Body Dose Commitment to Population of Eastern United States--Repository Breach by Meteorite Strike, man-rem

<u>Time of Event</u>	<u>Salt</u>	<u>Granite</u>	<u>Shale</u>	<u>Basalt</u>
Year of closure				
Spent Fuel	5.6×10^7	1.5×10^8	7.4×10^7	1.5×10^8
Reprocessing Wastes	5.2×10^7	4.4×10^7	3.2×10^7	5.2×10^7
Closure + 1000 Years				
Spent Fuel	1.0×10^7	2.7×10^7	1.3×10^7	2.7×10^7
Reprocessing Wastes	3.8×10^6	3.2×10^6	2.3×10^6	3.8×10^6
Closure + 100,000 Years				
Spent Fuel	1.8×10^5	4.8×10^5	2.4×10^5	4.8×10^5
Reprocessing Wastes	4.9×10^4	4.2×10^4	3.0×10^4	4.9×10^4
Closure + 1,000,000 Years				
Spent Fuel	6.3×10^4	1.7×10^5	8.5×10^4	1.7×10^5
Reprocessing Wastes	5.2×10^4	4.4×10^4	3.2×10^4	5.2×10^4

If a meteorite of the size described impacted anywhere in the nation, the area would probably be declared a disaster area regardless of whether or not it impacted over a waste repository. If a waste repository was nearby, monitoring teams could be dispatched to determine the levels of contamination in air, soils and water. Mitigating action would depend on the levels of activity found in various media and the areas involved. Action would range from withholding crops from use and moving dairy and beef animals to less contaminated areas, to removing contaminated soil where necessary and disposing of it under suitable controls.

The probability of a meteorite capable of striking the surface over the repository and producing a crater 2 km in diameter at the surface has been estimated to be 2×10^{-13} per year (Claiborne and Gera 1974). If the "mathematical expectation of societal risk" is taken as probability times consequence, the societal risk of death or serious genetic defect would

be from 4×10^{-3} to 3×10^{-2} health effects from the largest dose to the population as presented in Tables 5.5.3 to 5.5.5 over one million years. By way of perspective, in the United States the societal risk of death by lightning is about 120 per year, or about 1×10^8 deaths per million years (Accident Facts 1974). Thus, in this framework, the societal risk from a meteorite breach of a repository is about 3×10^{-10} that from lightning strikes. Even if the estimate of probability of this meteorite event was in error by a factor of a billion (as might be the case for the probability of a nuclear detonation over the repository), the risk to society remains less than that from lightning and can hardly be considered significant.

5.5.2 Breach of Repository by Fault, Fracture, and Flooding

This scenario is a combination of improbable events: first, a fracture or series of fractures either from the surface or from near an aquifer penetrates to the repository, second, the fractures are connected and permit water to reach the wastes. Two cases are presented, one where a fairly large stream of water penetrates the repository and leaches out radionuclides and then, following an assumed conduit, returns to the surface to form a stream. The second case presumes water reaches the repository and leaches out radionuclides and transports them beyond the boundaries of the host rock; some of the nuclides are then held up by adsorption on soils outside the repository area before slowly working their way to the biosphere. Such scenarios are presented as being independent of host rock properties.

These scenarios involve improbable combinations of events with very low probabilities of occurrence, and in some cases are contrary to the evidence available. For example, faulting of thick salt units does not generally lead to formation of permeable zones, and the plastic behavior of salt tends to heal any opening. Most of the known faults in salt formations confirm this self-healing behavior of salt (Claiborne and Gera 1974). Also, massive salt units generally occur in a geologic environment that contains clays, shales and argillaceous units that again tend to deform plastically. Faults in rock material that yield by brittle fracture (granite, basalt, some carbonates) are more likely to form permeable zones of crushed, broken rock than faults in salt. However, even in brittle rocks a fault zone may, through the grinding and crushing of the material, form a zone of very low to essentially no permeability. That any fault would form a continuously permeable conduit to the repository is doubtful, even if a fault should occur through the repository to the land surface.

In this scenario the repository is assumed to be breached by fracturing either at 1000, 10,000 or 1,000,000 years after repository closure. Water in the form of a stream of $2.8 \text{ m}^3/\text{sec}^{(a)}$ (100 cfs) invades the repository, flows among the wastes and enters the reference environment in the R river about 10 km from the repository center. The stream is assumed to be in contact with the wastes for one year. (This case simulates the subsequent

(a) Several comments were received on the draft Statement that such a large flow of water was unreasonable. However, the scenario is not all that unreasonable, at least in the long term. One can envision stream displacement as a result of ice dams, glaciation, or land slides to where the scenario becomes plausible at least to the extent of entry of water. Return of water to the biosphere is harder to imagine.

sealing of the breach line by further earth movement, healing because of the nature of the host rock or because of plugging of the water path by silt carried by the stream.)

Several studies have been performed to estimate the leach rate of waste by water. Two important factors affecting leach rate of a waste material are the waste form (chemical nature) and the temperature of the solid-liquid interaction zone. Data reported by Ross (1978), under repository conditions much more severe than would exist a thousand years after closure, indicate leach rates ranging from 10^{-8} to 10^{-5} gm/cm²-day for reactions between aqueous solutions and waste glasses in a devitrified and fractured state. Other studies by McCarthy et al. (1978), with conditions of 300°C and 300 atmospheres, have suggested changes in waste form properties which might lead to higher leach rates for some radionuclides in borosilicate glass. The same processes also caused recombination of some of the radionuclides with the immediate environment to a more stable form with a lower leach rate. Other studies in field situations at lower temperatures and pressure with the ground saturated with water have shown rates as low or lower than 10^{-10} gm/cm²-day for radionuclides in nepheline syenite glass (Merritt 1976). The leach rates used in consequence analyses, Table 5.5.6, are considered highly conservative in view of these studies and the likely temperature of the water contacting the waste.

TABLE 5.5.6. Estimated Leach Rates for Various Forms of Radioactive Wastes Used in Consequence Analyses.

<u>Waste Form and Assumed Geometry</u>	<u>Leach Rate gm/cm²-day</u>	<u>Number of Canisters Contacted</u>
High-level waste glass (assumed to be devitrified and fractured, and without any protection from the canister--1-cm cubes)	1 x 10 ⁻⁴ for first 10 days 1 x 10 ⁻⁵ thereafter	210
Spent fuel (1-cm-dia spheres)(a)	1 x 10 ⁻⁵	1230 PWR, 1320 BWR(b)
Fuel residue	1 x 10 ⁻⁵	30
Other TRU wastes	1 x 10 ⁻⁴	480, 560

(a) The fuel pellet simulating a combination of PWR and BWR fuels is taken to be a cylinder 1.16 cm in diameter by 1.16 cm long. Since the spent fuel dose calculations were made, the determination has been made that spent fuel may be fragmented following irradiation and that the area subject to leaching may be about 5 times that used in the original calculations (Pasupathi 1978). This factor has been applied to doses in this section.

(b) Subsequent to the calculations made for this Statement on the basis of 1230 PWR and 1320 BWR canisters (816-MTHM) contacted by water and subjected to leaching, the contents of the repositories in the various media were changed. The amounts of spent fuel contacted by water following a 12-m-wide fracture along the diagonal of a repository were estimated to be: salt; 340 MTHM, Granite; 870 MTHM, shale; 390 MTHM and basalt 810 MTHM (DOE/ET-0029). For all practical purposes the doses that follow would apply to the breach of granite and basalt repositories. Doses should be multiplied by a factor of 0.4 to obtain doses reflecting a breach in a salt repository and by a factor of 0.5 for a shale repository.

For dose calculations for spent fuel and vitrified high-level waste (the major contributor to dose from reprocessing wastes), doses may be calculated for other leach rates by multiplying the tabulated dose by the ratio of the assumed leach rates to the listed leach rate.

Seventy-year whole-body dose commitments have been calculated for the maximum individual using the data of Tables 4.4.3 and 9.3.34 in DOE/ET-0029, the methods described in Appendix D and the following assumptions. For cases in other than a salt repository, aquatic food is taken from, and recreational activities occur near, the $2.8 \text{ m}^3/\text{sec}$ stream of water from the repository (this assumption is perhaps overly simplistic since the stream flows for only one year and little time is available for an aquatic ecosystem to be established). Drinking water is taken from the river downstream from the point of contamination entry (the majority of the regional population resides down stream from the repository and the presumed point of entry of the stream). Contaminants in farm products and ground contamination doses were determined based on irrigation of land with water from the river. In the case of a repository in salt it was concluded that the $2.8 \text{ m}^3/\text{sec}$ effluent stream would be so laden with salt that no fresh-water biota would be present and that the maximum individual would derive his aquatic food from the river as opposed to the small stream.

Doses to the maximum individual are presented in Table 5.5.7. Population doses were also calculated on the basis of contamination of water in the R river. Seventy-year dose commitments to the maximum individual and the regional population were calculated for 1000, 100,000 and 1,000,000 years after closure of the repository.^(a) Doses to the regional population are presented in Table 5.5.8. Doses to other regions and for the breach in the year of repository closure may be found in DOE/ET-0029.

The range of population dose for the flooding and faulting event 1000 years after closure amounted to 8.8×10^4 to 1.7×10^5 for spent fuel and reprocessing wastes, respectively. Using the range of 100 to 800 health effects per million man-rem, the calculated total number of health effects attributable to this event, if it occurred as postulated, would be 9 to 140 depending on fuel cycle.

The probability of a fault intersecting the repository in a typical bedded salt basin such as the Delaware Basin has been estimated by Claiborne and Gera (1974) to be

TABLE 5.5.7. 70-Year Whole-Body Dose Commitment to Maximum Individual--
Repository Breach by Faulting and Flooding, rem

<u>Time of Event</u>	<u>Salt Media</u>	<u>Non-salt Media</u>
Closure + 1,000 Years		
Spent Fuel	3.0×10^{-1}	9.7
Reprocessing Waste	5.5×10^{-1}	1.5×10^1
Closure + 100,000 Years		
Spent Fuel	3.7×10^{-1}	8.6
Reprocessing waste	6.7×10^{-2}	1.5
Closure + 1,000,000 Years		
Spent Fuel	1.8×10^{-1}	4.3
Reprocessing waste	2.2×10^{-2}	4.5×10^{-1}

(a) Calculations were presented in the Draft DOE/EIS 0046-D for a stream breach in 2050. In deference to comments on the unreasonableness of this event, it is not presented here; detection would be almost certain and mitigation of affects possible. At 1000 years after closure the unrecognized contaminated stream does not seem unreasonable.

TABLE 5.5.8. 70-Year Whole-Body Dose Commitment to the Regional Population--Repository Breach by Faulting and Flooding

Time of Event	Man-rem
Closure + 1,000 Years	
Spent Fuel	8.8×10^4
Reprocessing waste	1.7×10^5
Closure + 1,000,000 Years	
Spent Fuel	$1.4 \times 10^{5(*)}$
Reprocessing waste	2.8×10^4
Closure + 1,000,000 Years	
Spent Fuel	7.1×10^4
Reprocessing waste	1.0×10^4

(*) The increase in dose between breaches at +1,000 and +100,000 years is due principally to the ingestion of ^{226}Ra from the decay chain of ^{238}Pu .

4×10^{-11} /yr. The frequency that a high pressure aquifer exists with canister and surface access is 0.005 (DOE/ET-0028, Sec. 7.4.9). A total probability for release to the biosphere is 2×10^{-13} per year.

Using the probability estimate of 2×10^{-13} /yr and the largest number of health effects calculated, 140 (Table 5.5.8), the mathematical expectation of societal risk would be at most 3×10^{-11} /yr or 3×10^{-7} health effects over 10,000 yr.^(a)

The population dose to the regional population from naturally occurring sources would amount to about 1.4×10^7 man-rem over the same time period. Even in the maximum case, that of 1.7×10^5 man-rem associated with release of radioactive material from nonsalt repositories, the doses are on the order of 1% of that from naturally occurring sources.

One of the potential long-term effects of release of radionuclides to the river would include the movement of these radionuclides to the ocean, where accumulation in mollusks may occur resulting in another pathway to human exposure. It was assumed that the following dilution factors^(b) were appropriate for concentrations of elements in an estuary; e.g., concentration of cobalt nuclides in estuary water would be 0.01 of their concentrations in the river.

(a) EPA commented that the calculation of probability was incorrect (see EPA ltr. comment #86; Vol. 3 App C. p 34). The EPA estimate of the probability of a faulting and water intrusion event was 4×10^{-7} over a 10,000-year period compared to 2×10^{-9} (2×10^{-13} /yr \times 1×10^4 yr) used in this Statement. EPA concluded that once a fault intersected the repository that the probability of water intrusion in the long term would likely be one. DOE believes the EPA argument has merit, however using the EPA figures increases the societal risk to only 6×10^{-5} over the 10,000 year period, which is still an insignificant societal risk.

(b) Dilution factors are highly dependent on the specific river system and estuary of interest. The dilution factors presented here were developed for movement of radionuclides from reactor effluent water at the Hanford Project in Eastern Washington via the Columbia River to Willapa Bay, Washington, where oysters are harvested.

<u>Element</u>	<u>Dilution Factor</u>	<u>Element</u>	<u>Dilution Factor</u>
H	2	Cs	100
C	2	Sm	100
Co	100	Eu	100
Ni	100	U	100
Sr	100	Np	100
Nb, Zr	100	Pu	100
Sb	2	Am	100
Sn	2	Cm	100
I	2		

Saltwater bioaccumulation factors were used to estimate the concentration of radionuclides in the edible portion of marine foods (Soldat, Robinson and Baker 1974). The 70-yr dose to the maximum individual from ingestion of mollusks (at a rate of 10 kg/yr) for repository breaches at 1,000, 100,000 and 1,000,000 years after repository closure were calculated. The largest of these, 7.2×10^{-2} rem to the whole-body, would add about 1% to the dose the individual would have received from naturally occurring sources for the same period and would not add significantly to the maximum individual's 70-year dose commitment.

The second scenario developed for the repository fracture and flooding assumes that radionuclides are leached from the waste and carried beyond the boundaries of the host rock and are then transported via moving (100 m/yr) ground water through the ground before entering the biosphere (the R river).

In this scenario a migration path length of 10 km was investigated, using sorption equilibrium constants (K_d values) measured or estimated under conditions at the Hanford Site, Richland, Washington. While these parameters are believed to be representative of average conditions to be expected at candidate sites, all factors could vary by several orders of magnitude.

Based on inventories of radionuclides in repositories and the models and dose calculation methods according to Lester et al. (1975) and Burkholder et al. (1975), doses were calculated for the maximum individual.^(a) Total body doses are presented in Table 5.5.9 as a function of time since disposal and for leach rates ranging from 0.1% to 0.01% of inventory per year.^(b)

The doses given in Table 5.5.9 were calculated to result from leaching of all wastes from a 50,000 MTHM example repository in salt. These doses would be about 2.5 times higher for the repositories in granite or basalt and about 1.3 times higher should the event occur in a shale repository due to larger amounts of waste contained in those repositories. In

(a) A computer model called GETOUT for hydrologic transport (Lester et al. 1978) was used in conjunction with a dose to biota model (Burkholder et al. 1975) as adjusted for parameters developed for the midwest reference environment.

(b) Several commenters on the draft concluded that total release of inventory in one year as presented in the draft Statement was out of the question. As a consequence the 100% removal per year case is omitted. The leach rates of 1×10^{-5} g/cm²-day used in the fracturing and stream flooding scenario amounts to about 1% of inventory removed per year, using assumptions that maximize the area available to contact water.

TABLE 5.5.9. 70-Yr(*) Accumulated Whole-Body Dose to Maximum Individual for Various Leach Rates and Times of Repository Breach by Fracturing and Ground-Water Intrusion (repository in salt--50,000 MTHM), rem

Years Since Disposal	Spent Fuel				Reprocessing Wastes			
	Breach 1000 years after closure, Leach Rate (yr ⁻¹)		Breach 100,000 years after closure, Leach Rate (yr ⁻¹)		Breach 1000 years after closure, Leach Rate (yr ⁻¹)		Breach 100,000 years after closure, Leach Rate (yr ⁻¹)	
	0.1%	0.01%	0.1%	0.01%	0.1%	0.01%	0.1%	0.01%
1.0 x 10 ³	5 x 10 ⁻²	5 x 10 ⁻³			5 x 10 ⁻²	5 x 10 ⁻³		
2.0 x 10 ³	1	1 x 10 ⁻¹			4 x 10 ⁻¹	4 x 1 ⁻²		
1.0 x 10 ⁴	5 x 10 ⁻¹	8 x 10 ⁻²			5 x 10 ⁻¹	5 x 10 ⁻²		
3.4 x 10 ⁴	5 x 10 ⁻²				5 x 10 ⁻²			
1.1 x 10 ⁵			2 x 10 ⁻²	4 x 10 ⁻³			3 x 10 ⁻²	3 x 10 ⁻²

(*) The computer program for this scenario used 50 rather than 70 years for exposure purposes. The values tabulated were adjusted upward for an additional 20-year exposure.

each case the host rock was assumed to be surrounded by a common soil-rock medium for which absorption rates would be the same.

The largest dose tabulated was 1 rem over 70 years if the event should occur. This is about one-seventh of the dose the individual would have received from naturally occurring sources and is believed to be of no consequence. The probability of this event occurring over a 10,000 year period is estimated to be in the neighborhood of 4×10^{-7} to 2×10^{-9} .^(a)

Over a time span of 100,000 years a peak dose occurs that is essentially independent of leach rate or time of repository breach. The dose is due principally to ²²⁶Ra, decay product of ²³⁸U (which has an extremely long half-life).^(b) At 1.4 million years after disposal the 70-yr dose to the maximum individual amounted to about 70 rem. This long-term radiological risk would not be significantly different from that of a natural ore body of similar content.

Doses to the regional population were not calculated directly for this scenario; rather, an estimate was made using a ratio of the per capita population whole-body dose and the whole-body dose to the maximum individual in the previously presented 2.8 m³/sec stream scenario. The ratio obtained was 1/5 and thus the per capita population dose was approximately one-fifth of the maximum individual dose. A whole-body dose to the regional population from ground-water contamination from breach of a 50,000 MTHM repository was estimated by multiplying the per capita dose by 2 million, the size of the regional population. Taking the largest maximum individual dose of 1 rem over 70 years to the whole body and using

(a) Probability of faulting over a 10,000 yr period of 4×10^{-7} was taken from EPA comment #113 on the draft to this statement. The probability of 2×10^{-9} over 10,000 years was developed from Claiborne and Gera (1974).

(b) About 10% of ²²⁶Ra is a result of decay of ²³⁸Pu produced in the reactor. About 90% of the ²²⁶Ra is from unaltered ²³⁸U in the fuel. After long periods of time, the principal source of potential dose to the public is the uranium from which the reactor fuel was made.

this conversion, a regional population dose of about 2×10^5 man-rem is obtained.^(a) By comparison the dose to this population from naturally occurring sources over the same period would be about 1.4×10^7 man-rem.

Unlike some of the other scenarios the contamination in this event could be expected to reach the environment continuously over a long period of time. For example, the 70-year dose to the maximum individual decreased from 1 rem to 0.5 rem between 2000 and 10,000 years after disposal (a factor of 2 would be lost in the imprecision of the estimate). The total dose to replicate regional populations over 10,000 years would be on the order of 3×10^7 man-rem (143-70 yr generations). The total regional population dose for this same period from naturally occurring sources would be about 2×10^9 man-rem. As noted earlier the probability of this event occurring is estimated to be between 4×10^{-11} and 2×10^{-13} /yr. The probability that it would occur sometime within a 10,000-yr period would be on the order of 4×10^{-7} to 2×10^{-9} . The mathematical expectation of societal risk would be less than one fatality over 10,000 years.

5.5.3 Faulting and Ground-water Intrusion to a Domestic Well

In this scenario a fault intersects a repository (non-salt) and water from an aquifer beneath the repository flows in small quantity through the repository to an overlying aquifer that is tapped by a domestic well. The domestic well is postulated to be located about 3 km down gradient from the fault and is capable of producing about 20 liters of potable water per minute.

In order to estimate the maximum consequences that might occur from the interaction with the buried waste, the assumption is made that all water flowing through the fault enters the domestic well. This suggests that the upper aquifer is of low permeability. Most domestic production wells are not drilled in aquifers of low permeability. Thus, for more usual permeabilities encountered, a much smaller fraction of the waste nuclides would arrive at the well. The water travel time from the fault in the repository to the domestic well would vary from 1000 to 2500 years depending on the streamline the water followed between the source and the well, while transport times for radionuclides could vary from a thousand to millions of years depending on the nature of the radionuclides and the sorption characteristics of the medium through which the water was flowing.

Doses were calculated from the rupture and leaching of 1320 BWR fuel assemblies and 1230 PWR fuel assemblies for the spent fuel repository; and 210 high-level waste canisters, 30 RH-TRU waste canisters and 480 barrels of RH-TRU waste for the reprocessing waste repository. All of the stated radioactive content is leached out over a 10,000-yr period.

The maximum 70-yr accumulated whole-body doses to the maximum individual from specific long-lived waste radionuclides that may be of interest and the time after connection with

(a) In reviewing the Draft EIS, EPA criticized this approach to population dose. At best, the method is a crude approximation of the population dose; but this approximation was made in lieu of reprogramming an existing dose code solely for this purpose. In any event the population dose could not exceed the dose of the maximum individual times the regional population (2×10^6 man-rem) and would likely be substantially less. (As in the previous scenario of a stream through repository, most of the population resides down stream from the entry of contaminated water.)

the repository that the dose would occur are as follows. Assuming that ^{129}I removed from dissolver off-gas is sent to the repository and is leached at roughly the same rate as from spent fuel, the doses are essentially the same for either fuel cycle option.

Radionuclide	Dose, rem	Time, yr
^{14}C	90	1×10^4
^{99}Tc	22	4×10^3
^{129}I	990	1×10^4
	(to the thyroid)	
^{135}Cs	0.2	1×10^6
^{237}Np	440	1×10^6

The probability of the event is estimated to be on the order of 4×10^{-7} to 2×10^{-9} over a 10,000-yr period.^(a)

Because of the extremely small probability of occurrence, and because of the very limited number of individuals that could be contaminated by such a well, the societal risk is believed to be insignificant.

5.5.4 Repository Breach by Drilling

In this scenario, about 1000 years after repository closure an individual (or group) drills 600 m into a waste repository in search of a mineral resource or for geologic study itself. Repository markers are no longer evident, are misunderstood, or are ignored. These individuals, while having the technology to drill to repository depth, do not possess or do not apply the knowledge and apparatus to assay material brought up in the drilling process and to discover its radioactive properties.^(b)

Because a probability for exploratory drilling could not be determined, an overall probability was not assigned to this event. In qualitative terms, someone could be exploring for potash, oil, etc.^(c) in the area of a repository in salt based on the same exploration principles that established the presence of the formation in the first place. In other formations such as granite, shale and basalt, associations with any particular resources are not as strong as in the case of salt. The probability that drilling will occur somewhere on the repository site is highly uncertain. If drilling occurs on the property, the

(a) Probability of faulting over a 10,000-yr period of 4×10^{-7} was taken from EPA comment #113 on the draft to this statement. The probability of 2×10^{-9} over 10,000 years was developed from Claiborne and Gera (1974).

(b) The drill crew may not be aware of radioactive material in the drilling mud as it is brought up; however, once samples are sent to their assay laboratory, the drillers would soon know of the radioactive nature of their exploratory effort. If the assay were crude they might conclude, in the case of drilling through a spent fuel element, that they had struck uranium, but very little sophistication in assay would be required to recognize that the radiation spectrum was not at all like that of natural uranium. The radiation characteristics of material brought up after passing through a solidified high-level waste canister would resemble natural ores even less.

(c) Because of the frequent occurrence of salt deposits at depths much shallower than 600 m the explorer would not likely be drilling to 600 m in search of salt.

probability that the drill (0.5 m in diameter) will strike a waste canister is 0.005 per drilling event, because of the relative cross sectional areas involved.

For dose calculations it is assumed that during drilling one-fourth of the waste in one canister is circulated to the surface with the drilling mud, and the radioactive material is uniformly distributed over 0.5 ha in the top 5 cm of the surface soil.

Table 5.5.10 lists the expected releases to air from contaminated surface soil. These values are based upon 1) a resuspension factor of 0.011/yr 2) the assumption that one-fourth of the radioactive material in the top 5 cm is available for resuspension and 3) that 0.10 of the material resuspended is respirable. The maximum individual is exposed, on the average, to the contaminated soil for 12 hr/day. Based on the releases given in Table 5.5.10 and methods of dose calculations presented in Appendix D, first-year doses and 70-yr doses to the maximum individual who will reside and grow crops for his consumption on the con-

TABLE 5.5.10. Respirable Radionuclides Released to the Atmosphere from Salt Repository Breach by Drilling 1000 Yrs After Repository Closure, Ci

Radionuclide	Spent Fuel	HLW
$^{14}\text{C}(\ast)$	1.7×10^{-4}	1.1×10^{-6}
^{126}Sb	1.2×10^{-5}	1.1×10^{-4}
^{126}Sn	1.2×10^{-5}	1.1×10^{-4}
$^{129}\text{I}(\ast)$	8.6×10^{-6}	7.4×10^{-7}
^{239}Np	3.4×10^{-4}	9.0×10^{-3}
^{240}Pu	1.1×10^{-2}	4.4×10^{-4}
^{241}Am	2.2×10^{-2}	3.2×10^{-2}
^{239}Pu	7.3×10^{-2}	6.3×10^{-4}

(*) The bulk of the C and I is volatilized during dissolution of the spent fuel and stored in separate containers and locations different than those used for HLW in the repository. For these two nuclides, 100% of the material resuspended is assumed respirable.

taminated land were calculated. The first-year whole-body doses amounted to 13 rem for drilling through a spent fuel canister and 19 rem for drilling through a HLW waste canister. The 70-yr whole body doses were 9.4×10^2 and 1.4×10^3 rem, respectively.

The predominant mode of exposure is direct radiation^(a) from contaminated soil and as a consequence, dose to the various organs is substantially the same the first year. During the 70-yr dose period the dose via the ingestion pathway increases substantially, particularly in terms of dose to bone. The 70 year accumulated doses as calculated might result in a small increase in risk of life shortening, contracting leukemia, etc.

(a) ^{241}Am is the principal contributor to the direct radiation dose. The dose from breach of a HLW canister was reported in the draft Statement, and in supporting documents, as about 100 times higher than here because an incorrect ^{241}Am inventory was used.

If the 0.5 ha of contaminated land were occupied by a housing project soon after the drilling incident with about 0.1 ha per lot, five families (probably about 25 individuals) might be exposed to the same extent as the maximum individual.

Seventy-year accumulated doses calculated for the regional population amounted to 1.1×10^2 man-rem in the case of spent fuel and 1.6×10^2 man-rem in the case of reprocessing wastes. All of the doses to the regional population (whose exposure would result principally from resuspension and air transport of radionuclides) are substantially less than those which would be received from naturally occurring radioactive sources (1.4×10^7 man-rem over the same period).

In the case of a repository in salt, the land (0.5 ha) would likely be contaminated with salt brought up with the drilling mud. As developed in more detail in DOE/ET-0029 the resulting ground contaminated by salt would not be well tolerated by ordinary crops.

Breach of a waste canister by exploratory drilling, if it occurred, could result in a small increase in risk of adverse health effects occurring among about two dozen people in the immediate area.

If exploratory drilling that reached the repository level were abandoned (whether a canister had been penetrated or not) it could provide a means of entry of water into the repository. It is believed that the bore hole would not remain open for long but if it did and significant quantities of water were to flow in and out the consequences would not reasonably exceed those described previously for faulting and flooding of a repository.

The key to mitigating action associated with a drilling accident is the discovery that radioactive material had been encountered. As stated, that knowledge would probably come from assay of the drill core or samples of the drilling mud. If the driller is aware that a drill has brought waste to the surface, standard decontamination methods could be used to recover the contaminants, dispose of them under suitable controls, and preclude essentially all of the previously mentioned radiological consequences.

5.5.5 Solution Mining

In this scenario a 47,000 MTHM example geologic repository in domed salt^(a) is breached by solution mining 1000 years after the repository is closed. Although this accident is typified by solution mining for salt recovery, solution mining is also used for extraction of other resources and for construction of underground storage cavities. This accidental breach of a repository is believed to be conceivable only for an industrialized society having technological capabilities substantially as exist today.

Basically, solution mining in domed salt involves drilling a well to the desired level and inserting a double-walled pipe so that water can be forced down the outer pipe into the salt, where it dissolves the salt into a brine and forces the brine back up through the center pipe (Kaufmann 1960). The life of such solution wells varies markedly, some failing in

(a) Solution mining of stratified salt is believed less likely than in dome salt because of less evidence suggesting the presence of salt.

a few years. For purposes of this accident analysis the well(s) could operate for 50 years before being abandoned because of failure caused by cave-in and crushing and plugging of piping with debris.

This accidental repository intrusion, as in the case of the drilling accident, is based on the assumption that repository markers are either no longer evident, are misunderstood or ignored. Salt deposits are relatively plentiful and drilling to 600 m for salt seems highly unrealistic. No probabilities could be assigned to this event; it is presented only as a hypothetical "what if" accident.

Ordinarily, once the brines are brought to the surface they are analyzed to determine the kinds and amounts of impurities such as calcium sulfate, calcium-magnesium carbonate, sulfides, etc., which would govern further processing to purify the salt. If radioactive waste is placed in repositories in salt formations, salt used for human consumption could be checked by radioanalysis as well (an institutionally administered precaution). Calculations suggest that radioactivity would be determinable with off-the-shelf gamma-ray spectrometer apparatus on samples of a few hundred grams at concentrations of waste in salt existing after a few days of mining operation and with certainty by one month of mining operation.

Assumptions of the scenario are that, although the salt stratum of the reference site is about 80 m thick, the salt removed is principally that from backfill, ceiling, pillars and floor where radioactive waste has been placed. In mining the repository about 33 million tons of salt would have been removed for waste placement. This represents about one-fourth of the total salt volume in the mined area (in the scenario, the repository has been backfilled completely with salt; actually backfill of about 60% is presently planned). The total salt postulated to be solution mined over 50 years is then about 130 million tons.^(a) This represents about 10% of the total salt contained in the salt stratum bounded by the repository area. If an equal amount of salt is mined in each of 50 years, the annual production would be about 2.6 million tons. In 1957 about 24 million tons of salt were produced in the United States (Kaufmann 1960). Such a solution mining operation for salt would exceed the size of those presently in operation in the United States; a very large operation in the United States produces about 0.4 million tons annually and in Europe a very large operation may produce on the order of 1 million tons of salt annually.

Given that 100 parts of water (at 20 to 100 C) by weight can dissolve 36 to 39 parts of salt, then over a 50-yr period a stream flow of 210 /sec is required to dissolve that much

(a) Although it is believed that radioanalysis of salt would result in termination of the operation soon after start-up, the scenario is developed based on removal of the repository salt over a 50-yr period. Amounts of wastes and salt brought to the surface over shorter periods of time are pro-rated based on water contact with all wastes by the end of 50 years. Consequences are based on the assumption that the presence of radioactivity goes undetected for one year.

salt. If an adequate source of water is available, nine wells each operating at about 23 /sec would be sufficient.

The actual solution chemistry of leached radionuclides moving into the salt brine is not known. An assumption of the scenario is that radionuclides leached from spent fuel mix completely with the salt brine and are carried to the surface. Although it may take 1/2 to 1-1/2 years to bring a brine well to production, in the scenario, the brine well produces immediately and continuously for 50 years, at the end of which the entire quantity of salt surrounding the waste would have been mined out. Water flow would follow a course of least resistance and would follow the previously mined cavern boundaries where possible; this maximizes the consequences.

Details of the calculations for leaching of radionuclides in spent fuel and in reprocessing waste with the disposed salt may be found in Sections 4.4 and 9.3, respectively, of DOE/ET-0029.

If 3% of the 2.4 million metric tons of salt mined per year is used for human consumption, then about 72,000 metric tons would be used for that purpose. If a person consumed 1800 g/yr then 72,000 metric tons of salt would provide for about 40 million persons. For purposes of this analysis the exposed population consists of 40 million persons.

Although daily monitoring controls on the salt would bring attention to the presence of contaminated salt, the possible failure of such monitoring was recognized. The producers' quality assurance laboratory may not recognize the failure for a week. That failing, it might take as much as a year before a consumer discovered the contamination. On this pessimistic series of circumstances the conclusion was that a reasonable upper bound on waste entering the food trade would be that in salt produced in one year. Therefore, the consequences of this accident in terms of radiation dose to an exposed population of 40 million persons from ingestion of contaminated salt for one year were calculated. The quantities of radionuclides which contributed significantly to whole-body dose and the doses are listed in Table 5.5.11.

TABLE 5.5.11. Amounts of Radionuclides (Ci) and 70-Year Whole-Body Dose (in rems to an individual) Resulting from Ingestion of 1800 g of Contaminated Salt 1000 years after Repository Closure

	Curies		rems	
	Spent Fuel Repository	Reprocessing Waste Repository	Spent Fuel Repository	Reprocessing Waste Repository
^{239}Pu	1.5×10^{-6}	2.2×10^{-8}	3.6×10^{-2}	5.5×10^{-4}
^{240}Pu	2.2×10^{-6}	1.2×10^{-7}	5.3×10^{-2}	3.1×10^{-3}
^{241}Am	4.5×10^{-6}	1.3×10^{-6}	3.0×10^{-1}	8.6×10^{-2}
^{243}Am	6.6×10^{-8}	2.4×10^{-7}	4.0×10^{-1}	1.5×10^{-1}
Total			3.9×10^{-1}	1.0×10^{-1}

The 70-yr whole-body dose commitment to the exposed population of 40 million persons would amount to 1.6×10^7 man-rem for such an event occurring in a spent fuel repository and to 4.0×10^6 man-rem from a similar event at a repository for reprocessing wastes. These

dose commitments are less than one-tenth of the 2.8×10^8 man-rem that the exposed population would receive over the same time period from naturally occurring sources. The relatively low population doses that might result if the event occurred indicates that the solution mining event would not constitute a significant societal risk.^(a)

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- (a) Other assessments of a solution mining event have been made in which different assumptions of repository size and amount of radionuclides reaching culinary salt were made. In particular the leaching was limited by the solubility of the uranium content of the waste. The contaminated salt was calculated to be distributed among 15 million persons. The 70-year dose to an individual for this event in a spent fuel repository amounted to 2.3 rem. This dose is about a factor of 6 higher than in the above analysis, but would also result in population doses less than those from naturally occurring sources.

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5.6 COST OF GEOLOGIC DISPOSAL

Constant dollar^(a) costs have been estimated for isolating both spent fuel and fuel reprocessing wastes in salt, granite, shale, and basalt formations. The costs include all construction, operating, and decommissioning costs. The costs of federal programs for repository research and development have not been included in the costs stated here, but are included in the systems cost estimates in Chapter 7. The cost estimates are stated in terms of constant 1978 dollars.

Results of this analysis show that for spent fuel repositories of constant size (800 ha), construction costs including mining and backfilling range from \$1 billion for bedded salt media to \$3 billion for basalt media. Total operating costs vary from \$590 million for a repository in salt to \$2.4 billion for one in basalt. However, since the allowable waste emplacement density in basalt is about 2.5 times greater than that in salt, unit costs for disposal in basalt are only about 70% higher than for disposal in salt. Costs of disposal in shale are similar to those in salt and costs of disposal in granite are similar to those in basalt. Cost estimates for reprocessing-waste repositories follow a similar pattern.

5.6.1 Construction Costs

The repository construction cost estimates include owner's costs as well as facility construction. Owner's costs include land acquisition, startup costs, owner's staff costs and other costs incurred by the owner--in this case the Federal government or its contractor--during construction. Facility construction costs are defined here to include the costs of all labor, equipment (including waste transport and emplacement equipment), buildings and structures, site improvements, shaft, corridor and room mining, backfilling, and architect/engineer services. Interest during construction is taken into account by discounting prestartup construction costs at 7% per year (constant dollar rate which excludes inflation). Construction cost estimates were generally based on designs prepared by the Office of Waste Isolation (OWI) in documents Y-OWI/TM-36, Vol. 1-23. These designs have been revised somewhat to reflect more efficient shaft design, construction and usage, revised mining schedules, increased surface storage of mined rock, and more workable surface handling facilities (see Vol. 4, Chapter 7 of DOE/ET-0028 (DOE 1979) or Section 5.3 of this Statement for repository descriptions). Construction costs are derived by estimating requirements for major equipment, buildings and structures, site improvements, and construction labor. These direct cost estimates are then factored to generate other direct costs, architect/engineer costs, and owner's costs.

The construction cost estimates, including a contingency factor, have an estimated accuracy range of +20%. This accuracy range reflects the uncertainties that are likely to be encountered during design and construction, but which are difficult or impossible to

(a) The term constant dollars means that the dollar value of the estimates in all future time periods is the same as the value of the dollar in the reference year (1978 in this statement); i.e., the effects of inflation are removed.

identify at this time, such as siting and engineering scope requirements necessary to provide a fully functional facility. Also included in the estimates are the possible variances of the assumed rock densities used in the development of mining costs. The contingency factors are such that, within the stated accuracy range, there is an approximately equal likelihood of the indicated cost overrun or underrun. The construction cost estimation methodology is explained in more detail in DOE/ET-0028, Vol. 1, Section 3.8.

Construction costs for repositories in different media are based on a fixed repository area of 800 ha (2000 acres). However, since waste emplacement density is a function of the thermal characteristics of each type of media, actual waste quantities emplaced differ for each 800 ha repository. Table 5.6.1 shows equivalent waste quantities emplaced, the resultant mining requirements and the construction costs. Operating costs and unit costs are also given in this table to facilitate comparisons of cost relationships. These costs are discussed in subsequent sections.

Since mining costs account for 30% to 50% of total construction costs, the total construction costs vary significantly between geologic media. However, emplacement capacity increases for media with higher mining costs (see Section 5.3) and the relative unit cost differences between geologic media are smaller than the relative construction cost differences. For example, construction costs for an 800-ha repository in basalt are about three times those of an 800-ha repository in salt for the once-through cycle, but the cost per

TABLE 5.6.1. Cost Estimates for 800-hectare Geologic Repositories

Waste Type	Geologic Media	Mined Quantity 10 ⁶ MT	Equivalent MTHM of Waste Emplaced		Construc- tion Cost ^(a) Millions of \$	Total Operating Cost ^(b) Millions of \$	Unit ^(c) Cost ^(e) \$/kg HM
			HLW ^(d)	TRU ^(d)			
Spent Fuel	Salt	30	51,000		1,000	590	52
	Granite	77	121,600		2,600	2,350	78
	Shale	35	64,500		1,300	810	57
	Basalt	90	121,600		3,100	2,390	87
Reprocessing Cycle Wastes	Salt	35	62,000	100,000	1,200	1,210	47
	Granite	53	69,000	108,000	2,000	1,940	77
	Shale	30	30,500	56,000	1,300	830	73
	Basalt	59	56,000	92,000	2,300	1,740	93

(a) Includes mining, backfilling and shaft sealing costs. Backfilling and shaft sealing costs are approximately 10% of total construction costs. Uncertainties in construction cost estimates are about +20%.

(b) Uncertainties in operating cost estimates are about +25%.

(c) Includes decommissioning costs. Uncertainties in unit cost estimates are about +50%.

(d) The metric ton of heavy metal (MTHM) equivalent of high-level waste stored at the initial repository is less than the MTHM equivalent of TRU wastes since the high-level waste must be cooled 5 years before it can be sent to the repository and deliveries to the repository lag behind TRU waste deliveries.

(e) Costs may be expressed in \$/GW-yr by multiplying by 38,000 KgHM/GW-yr.

kilogram of disposal in a basalt repository is only 67% higher. These unit cost relationships may change somewhat for repositories of optimized size at specific sites.

Construction costs for repositories in granite and basalt are much higher than for repositories in salt and shale, mainly because of mining cost differences. These differences arise because of different mined quantities, as noted previously, and because of higher unit mining costs reflecting the greater difficulty in hard-rock mining.

5.6.2 Operating Costs

Operating costs include the costs of direct labor, monitoring and safety, materials, utilities, maintenance, administrative and other overhead, hole drilling and/or trenching and retrievability sleeve placement. The materials category includes all overpacks, sleeves, and plugs used in the repository. Waste packaging or encapsulation costs were considered to be a predisposal cost and can be found in Section 4.9. Costs of the waste canisters are included in the encapsulation costs in the case of spent fuel or in the waste treatment and packaging costs in the case of reprocessing cycle wastes.

Labor, utilities, and maintenance requirements are based on estimates given in Y/OWI/TM-36, Vol. 10, 12, 14 and 16. Materials requirements, wage rates, and utility costs are based on annual receipts and price data described in DOE/ET-0028, Vol. 1, Section 3.8.2. Unit hole drilling, trenching, and sleeve placement costs were derived by the architect/engineer making the construction cost estimates and are detailed in DOE/ET-0028, Vol. 4, Sections 7.4.10.2 and 7.5.10.2. The allowances for maintenance, overhead, and miscellaneous costs have been derived by factoring either capital or direct labor costs. After inclusion of a 25% contingency factor the operating cost estimates have an estimated uncertainty of approximately $\pm 25\%$.

Total operating costs for the waste repositories are shown in the sixth column of Table 5.6.1. These figures represent the cumulative operating costs during the repository waste receiving periods. Cumulative operating cost differences between repositories are principally due to differences in amount of waste emplaced. The granite and basalt repositories generally have significantly higher cumulative operating costs than do repositories in salt and shale because of their greater capacity and longer operating lifetimes. Another significant factor in operating cost differences in spent fuel repositories is the higher cost of hole drilling in granite and basalt for canister placement.

5.6.3 Decommissioning Costs

Decommissioning costs are defined here to include decommissioning of the surface facilities and sealing of the repository shafts. Based on decommissioning cost estimates for other fuel cycle facilities, the decommissioning cost of the repository surface facilities is estimated at 10% of the construction cost of these facilities. Shaft sealing costs are estimated to be approximately \$25,000,000 per repository. The total decommissioning costs, excluding room backfilling, are shown in Table 5.6.2 for spent fuel and reprocessing-waste repositories.

TABLE 5.6.2. Decommissioning Costs for Spent Fuel and Reprocessing-Waste Repositories

Repository Media	millions of dollars	
	Spent Fuel Repository	Reprocessing-Waste Repository
Salt	50	55
Granite	50	54
Shale	50	54
Basalt	50	55

5.6.4 Unit Costs

Levelized unit costs are calculated charges per unit of production sufficient to recover all construction costs, including interest, and to pay all operating and decommissioning costs. For this study, the weighted cost of capital for the Federal government is assumed to be 7% but a range of 0 to 10% was utilized to develop uncertainty estimates. Additional information on the calculation of unit costs can be found in DOE/ET-0028, Vol. 1, Section 3.8.5.

The levelized unit costs for waste isolation in geologic repositories, based on the conceptual repositories used in this Statement, are shown in the last column of Table 5.6.1. These costs are expressed in dollars per kilogram of heavy metal of isolated spent fuel for spent fuel repositories or in dollars per kilogram of heavy metal reprocessed for reprocessing-waste repositories. Isolation in salt repositories costs significantly less than isolation in any other medium for either waste type with the exception of isolating spent fuel in shale. Shale is the next least expensive medium for disposing of either spent fuel or reprocessing cycle wastes. Granite is the next least expensive and basalt is the most expensive medium for isolating wastes. Unit cost differences between repositories storing spent fuel and repositories storing reprocessing waste (in the same geologic medium) do not appear to be significant, with the possible exception of repositories in shale. Because of the preliminary nature of the conceptual designs, uncertainty in the mining procedures and in the cost of money, the overall uncertainty in the total unit cost estimates is estimated to be +50%.

A breakdown of the unit costs for waste disposal by waste type for the reprocessing cycle wastes is shown in Table 5.6.3 for each of the four geologic media considered here.

TABLE 5.6.3. Unit Costs by Waste Type and Geologic Media

Waste Type	\$/kgHM			
	Salt	Granite	Shale	Basalt
HLW	24	39	41	51
RH-TRU Canisters	3	5	4	5
RH-TRU Drums	18	29	24	32
CH TRU	<u>2</u>	<u>4</u>	<u>4</u>	<u>5</u>
Total	47	77	73	93

5.6.5 Comparison with Other Cost Data

Recent repository cost estimates, including the estimates in this statement, use as their principal basis one of three independent repository conceptual design studies (Kaiser 1978, Stearns-Rogers 1979, OWI 1978). The Bechtel (1979) spent fuel disposal study uses the conceptual designs reported for Kaiser (1978) and Stearns-Rogers (1979) with variations based on differences in waste form. The repository cost included in DOE's preliminary spent fuel acceptance charge estimate DOE/ET-0055 (DOE 1978) is based on a planning study by Kaiser Engineers prior to the completion of their conceptual design estimates. The recent Environmental Impact Statement on Spent Fuel Policy, DOE/EIS-0015 (DOE 1980a), uses this same basis. The estimates in this Statement are based on OWI (1978) with design modifications as noted in Section 5.3.

The capital cost estimate for spent fuel repositories given in Bechtel (1979), DOE (1978), and DOE (1980a) is \$500 million with annual operating costs of about \$50 million. The main difference between these estimates and those in Table 5.6.1 is that a portion of the mining cost is allocated to operating cost instead of being totally included in the construction cost. The unit cost calculation for spent fuel disposal in a bedded salt repository of \$51/kg heavy metal in DOE/ET-0055 compares favorably with the \$52/kg calculated in Table 5.6.1 (both costs are in 1978 dollars).

In the DOE Statement of Position to the NRC Rulemaking Proceedings (1980b), cost estimates are given for spent fuel disposal in salt, granite and basalt media. Total capital, operating and decommissioning costs of \$2.2 billion (\$1.8 billion in 1978 dollars) for a bedded salt repository are in general agreement with this Statement. However, total costs for granite and basalt repositories reported in DOE (1980b) are about \$2 billion less than estimated here since the standardized mine layouts used in the DOE (1980b) estimate postulate substantially less rock removal per unit of waste emplaced than does this Statement.

5.6.6 Other Cost Considerations

Costs associated with the retrieval of spent fuel elements from the repository during the 5-year retrievable period, subsequent interim storage at the repository site and transportation to a new site are estimated to be no more than the figures presented in Table 5.6.4.

TABLE 5.6.4. Spent Fuel Retrieval Costs

	\$/kgHM			
	<u>Salt</u>	<u>Granite</u>	<u>Shale</u>	<u>Basalt</u>
Retrieval	14	18	15	18
Interim Storage	22	22	22	22
Shipment to New Repository (~1500 mi)	32	32	32	32
Total	68	72	69	72

The disposal costs given in Table 5.6.1 apply for all cases in which spent fuel disposal requirements are at least equivalent to 48,000 MTHM. For the case in which disposal requirements are limited to the 1980 inventory of spent fuel (about 10,000 MTHM), unit repository costs would be approximately:

	<u>\$/kgHM</u>
Salt	90
Granite	100
Shale	90
Basalt	110

The total costs of waste management including disposal are presented and compared to the total cost of electric power production in Section 7.6.

REFERENCES FOR SECTION 5.6

- Bechtel National, Inc. 1979. An Assessment of LWR Spent Fuel Disposal Options, Vol. 1 and 2, ONWI-39, Office of Nuclear Waste Isolation, Columbus, Ohio.
- Kaiser Engineers Inc. 1978. Conceptual Design Report--National Waste Terminal Storage Repository in a Bedded Salt Formation for Spent Unreprocessed Fuel, DOE Contract No. EY-77-C-05-5366. U.S. Department of Energy, Washington, D.C.
- Office of Waste Isolation. 1978. Technical Support for GEIS: Radioactive Waste Isolation in Geologic Formations, Y/OWI/TM-36/1-23, Union Carbide Corporation, Oak Ridge, Tennessee.
- Stearns-Rogers Engineering Co. 1979. Conceptual Design Report--National Waste Terminal Storage Repository for Storing Reprocessing Wastes in a Dome Salt Formation, DOE Contract No. EY-77-C-05-5367.
- U.S. Department of Energy. 1978. Preliminary Estimates of the Charge for Spent Fuel Storage and Disposal Services, DOE/ET-0055, Washington, D.C.
- U.S. Department of Energy. 1979. Technology for Commercial Radioactive Waste Management. DOE/ET-0028 (5 volumes), Washington, D.C.
- U.S. Department of Energy. 1980a. Final Environmental Impact Statement--U.S. Spent Fuel Policy, DOE/EIS-0015, Vol. 4, Washington, D.C.
- U.S. Department of Energy. 1980b. Proposed Rulemaking on the Storage and Disposal of Nuclear Waste (Confidence Rulemaking)--Statement of Position of the United States Department of Energy, DOE/NE-0007, Washington, D.C.

5.7 SAFEGUARDS INCLUDING PHYSICAL PROTECTION FOR GEOLOGICAL DISPOSAL

Facilities associated with geologic repositories will employ safeguards and physical protection measures commensurate with the potential risks to society associated with the waste material (see discussion in Section 4.10), and the surface facilities at these sites would receive the principal emphasis. After emplacement in the geologic repository, the spent fuel and wastes would be very inaccessible for theft or diversion. Sabotage, if achieved, would have a minimum effect on the safety and health of the public because of the containment of the waste in a solid material that is difficult to pulverize and disperse. Nevertheless, sabotage of the facility and the waste packages must be guarded against until repository closure.

5.7.1 Geologic Disposal of Spent Fuel

Safeguards, including physical protection measures afforded vital material, would be required for the spent fuel elements as they are received, inspected, and made ready for geologic placement. This material is not attractive for theft or sabotage for the reasons given previously (Section 4.10.1.2), and in addition it becomes more inaccessible at this facility. Moreover, the currently required physical protection measures include controlled access through two barriers plus an adequate security force, and a contingency plan (response force) as required by the Federal regulations (10 CFR 73). Records of waste disposition to provide traceability from origin to final disposal will be maintained (43 CFR 195 1978).

After emplacement and closure in the geologic repository, the spent fuel would be essentially inaccessible for sabotage or theft. A successful intrusion and theft of HLW containers or sabotage in place would be unlikely because of the limited access to the containers, the operational control over entry, and the physical security provided at the access points in the surface facility. After repository closure the waste would be available only through re-excavation or mining. Theft or sabotage after closure and decommissioning does not appear credible because the effort would be readily detectable.

5.7.2 Geologic Disposal of Solidified High-Level Waste and Transuranic Wastes

The physical protection required for the surface facility handling these wastes includes measures to protect the facility and material from intrusion, theft and sabotage. These measures would be similar to those in any facility handling moderately hazardous material. At the repository these materials would be quite inaccessible to the public, and in a form that is not attractive for theft or sabotage. The solidified high-level waste would be too radioactive for adversaries to handle except remotely behind heavy shielding which, as shown in earlier discussions, makes this material inherently unattractive. Routine accountability programs would record the transfer of this material to its geologic disposal location. After geologic emplacement this material would be relatively inaccessible for theft. Sabotage, if ever attempted, would have little affect on the public because of the containment of the waste. After closure, theft or sabotage does not appear credible because mining or re-excavation would be required to gain access. Such an operation would be difficult to conceal and could be easily prevented.

REFERENCES FOR SECTION 5.7

Code of Federal Regulations. Title 10, part 73.

"Storage of Spent Fuel in an Independent Spent Fuel Storage Installation." 1978. Federal Register, Vol. 43, p 195.

5.8 IRREVERSIBLE AND IRRETRIEVABLE COMMITMENT OF RESOURCES ASSOCIATED WITH GEOLOGIC REPOSITORIES

Resources that will be irretrievably committed in disposal of radioactive wastes in geologic repositories are the energy resources consumed in repository construction and operation, cement (a relatively energy intensive material in concrete) and any canister or engineered barrier materials committed to the repository with the waste. Ranges of commitments of these resources for the several geologic disposal media, on a normalized energy production basis of one GWe-yr, are presented below:

	<u>Spent Fuel Repositories</u>	<u>Fuel Reprocessing Waste Repositories</u>	<u>Approximate U.S. Annual Production</u>
Propane, m ³	1.6 - 1.9	1.4 - 3.1	1 x 10 ⁶
Diesel Fuel, m ³	1.1 x 10 ² - 1.8 x 10 ²	1.6 x 10 ² - 2.2 x 10 ²	4 x 10 ⁸
Gasoline, m ³	1.2 x 10 ¹ - 1.4 x 10 ¹	1.1 x 10 ¹ - 2.2 x 10 ¹	6 x 10 ⁸
Electricity, kw-hrs	9.9 x 10 ⁵ - 1.3 x 10 ⁶	1.2 x 10 ⁶ - 1.8 x 10 ⁶	2 x 10 ¹²
Manpower, man-yrs	2.1 x 10 ⁴ - 2.9 x 10 ⁴	3.2 x 10 ⁴ - 5.3 x 10 ⁴	4 x 10 ⁶ (a)
Steel, MT	1.8 x 10 ¹ - 2.8 x 10 ¹	6.1 x 10 ¹ - 8.1 x 10 ¹	1 x 10 ⁸
Cement, MT	2.1 x 10 ¹ - 2.7 x 10 ¹	3.1 x 10 ¹ - 4.4 x 10 ¹	7 x 10 ⁷
Lumber, m ³	1.8 - 2.1	2.4 - 3.3	3 x 10 ⁹

(a) Construction and mining.

Even at an installed nuclear power capacity of 250 GWe operating over several decades the above material and energy commitments are but a small fraction of that used for the total economy. To give additional perspective to the consumption of energy, fossil fuels, and electrical consumption, each were converted to units of energy expended in deep geologic disposal of waste per unit of energy produced by the fuel from which the waste came. In the case of spent fuel 0.04% of the energy produced was consumed in geologic waste disposal and in the case of fuel reprocessing wastes 0.05% of the energy produced was consumed. On the above bases it is concluded that the irretrievable commitment of the above materials is warranted.

5.9 SHORT-TERM USES OF THE ENVIRONMENT VERSUS LONG-TERM PRODUCTIVITY

In terms of short-term use, about 800 ha (2000 acres) will be restricted from present use and until the repository is decommissioned (on the order of 30 years). After decommissioning, this land could be returned to its former use. An exception would be the area on which excess rock had been stockpiled, assuming no use elsewhere had been found for the rock. The area that this rock would cover would depend on the height to which it was finally piled. Characteristics of specific sites would probably dictate the size and shape of the rock storage pile(s). If the height of the storage pile were about 3 m (10 ft) the pile (ignoring the angle of repose of the rock) would occupy an area about 2200 m (1.4 miles) on a side. If left in this state, this large pile would constitute a cost in terms of lost productivity of the surface soils and in terms of an aesthetically displeasing visual impact. On the other hand, this large pile for granite and basalt repositories could be moved and modified to form a suitable marker for the repository. The costs would be balanced by the benefits of permanent isolation of radioactive waste far beneath the earth's surface.

5.10 UNAVOIDABLE ADVERSE ENVIRONMENTAL IMPACTS ASSOCIATED WITH RADIOACTIVE WASTE DISPOSAL IN GEOLOGIC REPOSITORIES

Impacts associated with nonradiological accidents during construction of geologic repositories and the dose to the work force emplacing the wastes, are perhaps the most significant unavoidable adverse impacts. In the strictest sense, such accidents should be avoidable, but experience in construction and mining suggests they will happen even with the best safety programs. The estimated number of expected fatalities (or permanent disabling injuries) ranged from 6 to 17 per 1000 GWe-yr of electrical energy generation, depending on repository media and whether disposal is for spent fuel or for fuel reprocessing wastes. While the number of lives which might be lost during mining operations could be obviated by some other disposal alternative, the radiation dose from waste disposal would be comparable (at least at this stage of estimating) for alternative disposal methods. (As a point of perspective, about 200 linemen would be expected to lose their lives in the process of bringing 1000 GWe-yr of electrical energy to its users regardless of the generation mechanism.)

The radiation dose to the work force emplacing the waste was estimated to be 4×10^3 man-rem for spent fuel and 8×10^4 man-rem for fuel reprocessing wastes for 1000 GWe-yr of electrical energy production. Using the conversion of 50 to 500 fatal cancers per million man-rem, about 2 radiation-related fatalities would be expected for emplacement of spent fuel; and 4 to 40 from emplacement of fuel reprocessing wastes for 1000 GWe-yr.

Radiation dose to population groups was not significant even in the case of postulated accidents during repository operation. Hazards to workers from potential operational accidents (canister drop down the mine shaft) were found to be very serious; however, additional safety features as suggested would reduce the risk substantially. For disruptive events in the long term the societal risk from wastes disposed of in geologic repositories was found to be small in comparison to societal risks such as from lightning strikes.

Adverse impacts on the terrestrial and aquatic environments could result from inadequate precautions taken for management of mined rock stockpiled on the surface, particularly in the case of repositories in salt and to a lesser extent in the case of repositories in shale.

The potential for boom/bust socioeconomic problems was determined to be very high for sites that may be isolated from needed labor pools. Although highly site specific, plans to lessen or obviate socioeconomic impacts are likely to be required for the site selection process.

There will likely be adverse psychological impacts among some members of the public because of the presence of a repository in their locality. A program to explain the exact nature of the repository facility and the multiple features present to prevent release of radioactive materials could lessen the concerns of the local public as long as information is completely presented and the activities of DOE are open to the scrutiny of local community leaders.

Chapter 6

ALTERNATIVE CONCEPTS FOR WASTE DISPOSAL

A number of possible alternative methods for the disposal of nuclear waste have been suggested. These concepts have been evaluated and developed to various degrees by different organizations. The status of technology is described in this section, as are advantages and disadvantages of each concept. The intent is to address the various concepts as consistently as possible to facilitate the comparison of the potential impacts of their implementation.

The alternative concepts discussed are: the very deep hole, rock melting, island repository, subseabed, ice sheet, well injection, transmutation, and space. These are all compared to the mined repository concept.

6.1 PRESENTATION/ANALYSIS OF ALTERNATIVE DISPOSAL CONCEPTS

This section presents concept descriptions and discussions of potential health and environmental impacts for eight radioactive waste disposal methods that have been suggested as alternatives to disposal in mined geologic repositories. These presentations are based on sections from the draft of this Statement, updated to incorporate current information resulting from continuing development and evaluation of alternative concepts. Information presented here is taken from available results of relevant studies. References, cited throughout the text to indicate sources of significant parameter values and statements, are listed at the end of subsection 6.1. In addition, bibliographies are provided in Appendix M to indicate other information sources for each concept. The concept descriptions are also supported by information in Chapters 3, 4, and 5 of this EIS and reference is made to those chapters as appropriate.

The discussion of each disposal concept covers the following topics:

- Concept Summary
- System and Facility Description
- Status of Technical Development and R&D Needs
- Impacts, Both Preplacement and Postplacement
- Cost Analysis
- Safeguard Requirements.

Because concept descriptions, environmental impacts, and estimated costs for each option were taken from various sources that used different basic assumptions, the information provided here for each concept is not normalized to a standard set of conditions, e.g., a common

annual throughput or a common environment. As an example, the well injection concept section presents radiological impact information extracted from a reference which addresses the impacts of intermediate level waste disposal. This is done to provide the reader with related information that may be important to the understanding of the concept. In addition, the space disposal and transmutation concepts require chemical processing of spent fuel to prepare waste for disposal or elimination. Accordingly, comparisons between these concepts and, for example, others not requiring processing would be difficult. For instance, transportation costs in the processing case could not be compared with those for disposal of spent fuel.

Four of the concepts (very deep hole, rock melt, space, and subseabed), however, were covered in a common reference and thus have a common basis. The other options are not normalized because, for example, while linear extrapolation to a higher or lower quantity of waste handled may result in a more or less conservative estimate of impacts and costs for a particular option, it may also bias any comparative analysis for or against that concept. Also, the descriptions, impacts, and costs that have been reported for some of the alternatives are incomplete because of the early stage of the alternatives' technical development.

In addition to being, in many cases, incomplete, the cost and impact data should be considered speculative. For example, the costs projected for the development of an alternative are generally based on judgment regarding the current state of technical uncertainty for the alternative. In practice, many such cost estimates do not adequately anticipate the expanded scope of activities that may result as additional uncertainties and issues are identified in attempts to resolve the original set of uncertainties. It was felt, therefore, that manipulating costs and impact information may indicate more significance than is warranted.

The disposal methods along with rates used as a basis for defining each of the alternatives, including the mined geologic repository, are:

<u>Alternative</u>	<u>Disposal Rate, MTHM/yr</u>	<u>Reference</u>
Mined Geologic Repository	6,000	Chapter 3
Very Deep Hole	5,000	Bechtel (1979a)
Rock Melt	5,000	Bechtel (1979a)
Island	Disposal rates similar to mined geologic repository. Ocean transportation similar to sub-seabed concept, see section 6.1.	Chapter 5, and Section 6.1.4
Subseabed	5,000	Bechtel (1979a)
Ice Sheet	3,000	MITRE (1979)
Well Injection	Unspecified	ORNL TM 1533, DOE (1979)
Transmutation	2,000	Blomeke et al. (1980)
Space	5,000	Bechtel (1979a)

Frequently, numbers taken from the various references are rounded to an appropriate number of significant digits in an effort to simplify this section of the document.

The general approach to each of the topical discussions used to describe the alternatives is as follows.

Concept Summary. The concept summary provided for each alternative contains a general discussion of the disposal concept, highlights significant technical aspects of the concept, and establishes a basis for specific system and facility descriptions, technology status, and environmental impact analyses.

System and Facility Description. In this section, the systems and facilities associated with a reference repository system design for each alternative disposal concept are described. Each description begins with a discussion of the fuel-cycle options reflected in the reference system design. The options and the selections made are illustrated by a standard diagram.

The waste-type compatibility for each concept is discussed, providing a basis for defining waste types that can and cannot be accepted by the disposal system. This section also indicates if the total fuel cycle involves chemical processing and if there is a need for a mined geologic repository (or other additional facility) to accept some portion of the waste.

The waste management system descriptions cover predisposal treatment and packaging (with reference to Chapter 4), surface facilities and equipment, and transportation systems. These descriptions vary substantially because of differences among the alternatives, e.g., space disposal compared to transmutation. System descriptions provide a basis for subsequent discussion of technology status and R&D requirements, potential environmental impacts, and cost analysis.

Status of Technical Development and R&D Needs. This section provides an insight into the technical status and R&D needs associated with the development of each disposal option. The discussions are based on the most current reports contained in the large body of references available for disposal options. Emphasis was placed on documents prepared by organizations that have played a definitive role in the development or evaluation of specific options.

Each disposal option is at a different stage of development ranging from ice sheet and rock melt, which are in only the early conceptual stage, to well injection, which has been used for the disposal of remotely handled waste at the Oak Ridge National Laboratory. Wide disparity in the states of development, however, should not be used to connote the degree of difficulty anticipated in deploying a particular option.

Current technological issues unique to each option are identified. These issues depend on the state of development. As knowledge is accumulated and refined on a specific concept to resolve technical issues, it may often reveal additional technological concerns to be resolved.

Specific research and development requirements ascribed to each disposal option are those contained in references provided by organizations involved in the development or evaluation of the particular disposal option. The requirements identified are based on technological issues and programmatic needs.

Estimates for implementation time and research and development costs depend on the degree of planning information available for the disposal concept. For example: no estimates

are identified for well injection because of lack of definitive program plans. Available estimates for space disposal go through concept definition and evaluation only. Estimates for ice sheet disposal, however, include all of the currently anticipated activities required to develop and implement an operational system.

Impacts. Impacts are presented on the basis of information found in the reference material. Impacts for these sections are separated into Health Effects Impacts (the human environment) and Natural System Impacts. Natural System Impacts include impacts to ecological and geologic/hydrologic systems. The term Natural System Impacts therefore includes impacts other than those to the human environment. The reader is cautioned that for those alternatives that are more advanced in their technical development, a greater number of environmental impacts are identified. Likewise, for those disposal methods that are in a preliminary stage of development, there may be other environmental impacts that have not yet been determined.

In general, the methodology followed in calculating impacts is not described, but reference is made to original material where the reader can find this information. No attempt has been made to develop a common impact assessment methodology, so the methods applied vary from study to study. For these reasons, the values presented are not always comparable on a one-to-one basis. It is believed, however, that sufficient information is provided to allow a qualitative comparison of the alternatives (see Section 6.2).

Cost Analysis. The cost analyses provide capital, operational, and decommissioning cost estimates based on information available from references authored by organizations involved in the evaluation or development of the specific disposal options. The costs are those directly attributable to the disposal mode under consideration and not on support modes such as waste preparation or routine transportation. All cost estimates are given in 1978 dollars, derived by an adjustment of 10 percent per year of estimates based on non-basis years.

The reader is cautioned about the preliminary nature of the cost estimates. Also, in many cases, due to the underdeveloped status of most of the alternatives, full cost data are not available. In such cases only referencable information is presented. No attempt is made to estimate system or component development, capital, operating or decommissioning cost where these, estimates could not be based on open literature reference. For example, in the case of the transmutation concept, a comprehensive and conclusive fuel cycle cost analysis has not been performed such that an aggregate cost estimate could be prepared. In addition, the impacts to the costs of disposal of the residual wastes from the transmutation concept are not known.

The estimates do not include transportation and waste-form preparation costs associated with the disposal method. However, unique transportation and waste-form requirements, in addition to the need for supplemental storage or disposal, are identified.

The cost analyses for very deep hole, seabed, rock melt, and space disposal are based on estimates contained in a current reference that used consistent waste disposal rates over the same time period. The available costs for the other disposal options, including the

mined geologic repository, are not normalized to the same waste disposal scenario. Cost estimates are sufficiently accurate, however, for a qualitative comparison.

Safeguard Requirements. In this section, the vulnerability of each alternative waste disposal concept to the diversion of sensitive materials or terrorist acts of sabotage is qualitatively discussed. In addition, the features unique to the alternative that enhance or detract from that vulnerability are described. For more detailed discussion of safeguards for predisposal operations the reader is referred to Section 4.10.

6.1.1 Very Deep Hole

6.1.1.1 Concept Summary

The very deep hole (VDH) concept involves the placement of nuclear waste as much as 10,000 m (32,800 ft) underground, in rock formations of high strength and low permeability. In this environment, the wastes might be effectively contained by the distance from the biosphere and the location below circulating groundwater as they decay to innocuous levels (OWI 1978 and ERDA 1978). To act as a nuclear waste repository, the host rock would have to remain sealed and structurally stable under the heat and radiation introduced by the wastes. Potential rock types for a repository of this kind include crystalline and sedimentary rocks located in areas of tectonic and seismic stability.

An immediate question concerning this concept is: "How deep is deep enough?" The required depths would place the wastes far enough below circulating ground waters that, even if a connection develops, transport of materials from the repository to the surface would take long enough to ensure that little or no radioactive material reaches the biosphere (LBL 1979). The absolute value of this depth is not yet determined.

Defining the necessary depth at a given site requires determining site-specific limits on the transport of radioactive materials to the biosphere, the site-specific hydrologic regime, and the heat-source configuration (waste packing). Available data from the literature, primarily from the oil and gas industry, show that some sedimentary rocks are porous and permeable and may contain circulating groundwater to depths in excess of 9,000 m (30,000 ft). Investigations of crystalline rock, although very limited, suggest that at much shallower depths some such rocks have relatively low porosities and permeabilities. Hence "very deep" for these crystalline rocks may mean just a few thousand meters instead of the 9,000 m or more required for sedimentary rocks. Once the required depth has been determined, the technology for making the hole to that depth and the ability of the surroundings to accept the heat source become the limiting factors. It is clear that problems of making the hole, holding it open, emplacing the waste, and sealing the hole must be considered together. Should shallow depths be determined as adequate, many of the potential problems of the very deep hole concept (e.g., drilling technology and ambient conditions at depth) would be mitigated.

The concept assumes that disposal in very deep holes would not permit retrieval of wastes. It would also provide assurance that no climatic or surface change will affect disposal.

Environmental impact considerations for the very deep hole concept are those associated with drilling a deep well or sinking a deep shaft, constructing the predisposal surface facilities, emplacing the wastes, decommissioning the facilities, and ensuring long-term containment of the wastes.

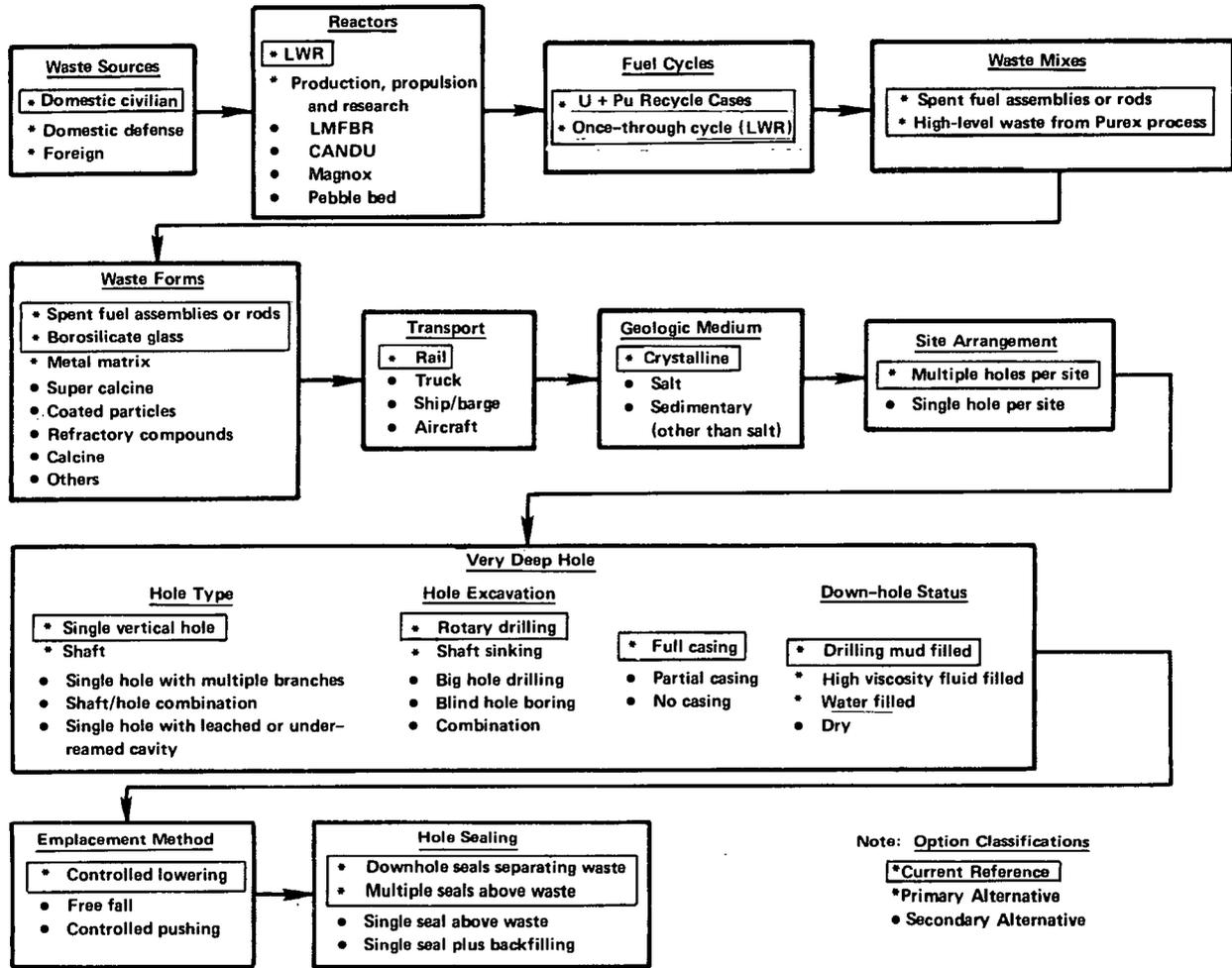
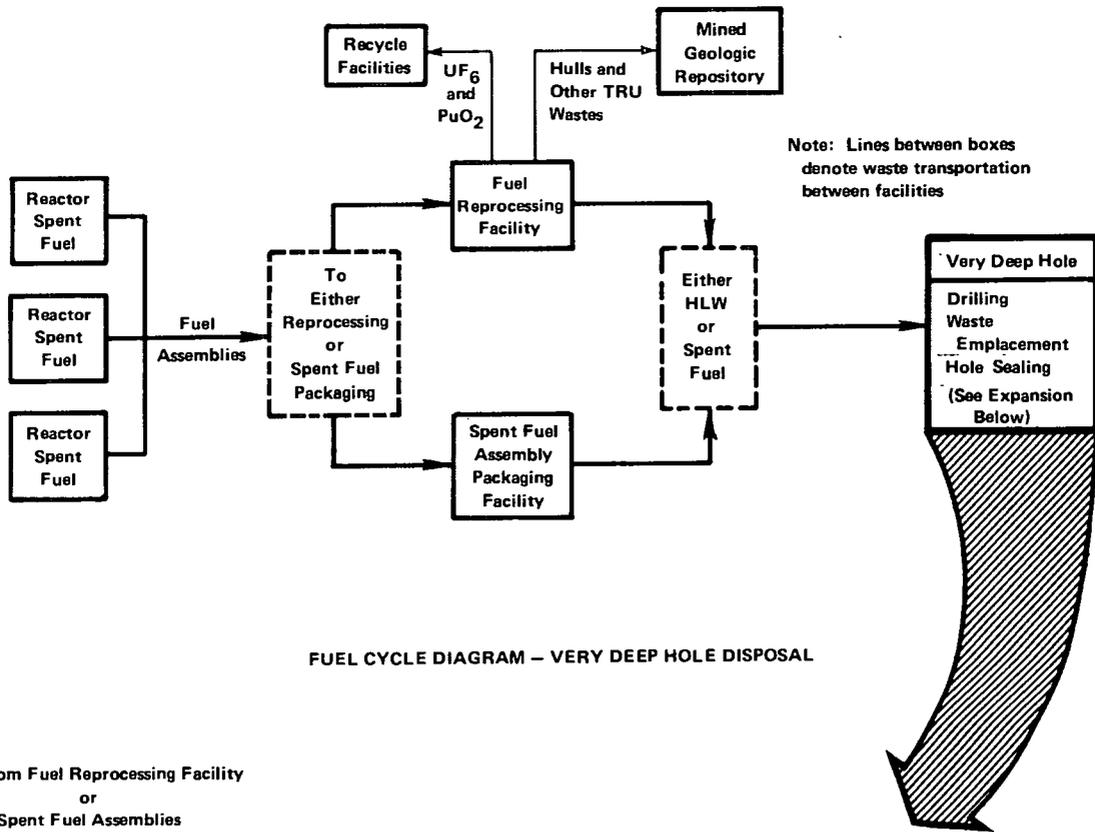
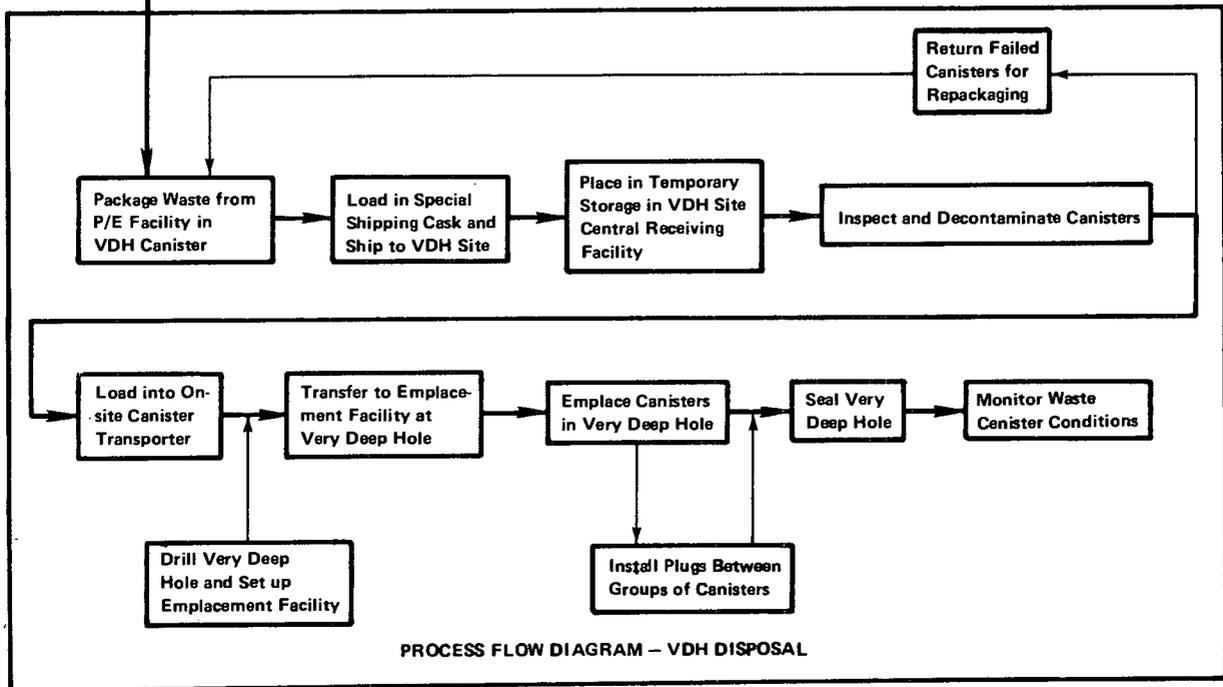


FIGURE 6.1.1. Major Options for Very Deep Hole Disposal of Nuclear Waste



FUEL CYCLE DIAGRAM - VERY DEEP HOLE DISPOSAL

HLW from Fuel Reprocessing Facility
or
Spent Fuel Assemblies



PROCESS FLOW DIAGRAM - VDH DISPOSAL

FIGURE 6.1.2. Waste Management System--VDH Disposal

6.1.1.2 System and Facility Description

System Options

The reference concept for the initial VDH disposal of nuclear waste has been developed from a number of options available at each step from the reactor to disposal in the very deep hole.

Various options to be considered for VDH disposal are summarized in Figure 6.1.1. The bases for selection of options for the reference concept (those blocked off) are reviewed in detail in various documents listed in Appendix M.

Because options for the waste disposal steps from the reactor up to, but not including, the geologic medium are similar for mined geologic repositories and VDH disposal, the options selected for the reference design are similar for the two concepts. From that point on, the options selected for the reference design are based on current program documentation for VDH disposal.

Waste-Type Compatibility

Very deep hole disposal would be limited to unprocessed spent fuel rods and the HLW from uranium-plutonium recycle cases. Because of cost constraints, VDH disposal of contact handled and remotely handled TRU wastes is not considered likely. Handling the large volume of these wastes would substantially increase drilling activities, costs, and the extent of adverse environmental impacts for VDH disposal. Thus, the low- and intermediate-level TRU wastes would require some other form of terrestrial disposal. It is assumed for the reference case that these wastes would be placed in mined geologic repositories.

Waste-System Description

The reference concept design was selected through judgment of a "most likely" approach based on available information and data. The fuel cycle and process flow for the reference concept are shown in Figure 6.1.2. In the reference concept, a VDH repository is designed for disposal of 10,200 canisters per year of spent fuel or for 2,380 canisters per year of solidified HLW. With a 40-year repository operation period, emplacement of spent fuel would require 68 holes per year with 150 canisters placed in each. Multiple holes would be drilled while others are being filled. HLW would require emplacement of 375 canisters per hole in six to seven holes per year (Bechtel 1979a), also with simultaneous drilling and emplacement operations.

Predisposal Treatment and Packaging. The predisposal treatment of waste for the VDH concept would be identical in many respects to the predisposal treatment of waste for the mined geologic repository concept. Chapter 4 of this document discusses the predisposal systems for both spent fuel and HLW common to all of the disposal concept alternatives.

The specific waste form required for emplacement in the deep hole is not yet identified. The waste form and canister would have to be structurally strong to resist downhole stresses and crushing forces, and chemically resistant to the waste emplacement medium. A metallic matrix or a granular waste form would be possible (Bechtel 1979a).

The canister would have to provide for safe handling, shipping, and emplacement of the waste. Both the HLW and the spent fuel canisters would have to be packed solidly to avoid crushing due to hydrostatic pressure of drilling "mud" (lubricant) left in the hole to counter lithostatic pressure. The canisters and spacers would have to be dense enough to sink through the mud slurry to the bottom of the hole. Carbon steel is considered as one candidate canister material that will fulfill these requirements (Bechtel 1979a). However, more complex designs using multiple barriers may be required.

The canisters for both HLW and spent fuel would have to be small enough for emplacement in a hole lined with a steel casing. HLW canister dimensions identified for the reference case accommodate the fuel. Dimensions identified for the reference case are 36 cm (14 in.) diameter and 4.6 m (16 ft) long (Bechtel 1979a and TID 1978).

Site. The critical geologic parameters that will determine the feasibility and impact of nuclear waste disposal in a deep hole system and that must be considered in site selection are:

- Lithology
- Tectonics and structural setting
- Hydrologic conditions
- States of stress
- Mechanical properties of the rocks at depth
- Natural thermal regime
- Geochemical reactions.

The interactions of these parameters and the effect of heating by the waste (thermomechanical factors) may also be significant. Geologic assumptions underlying the VDH concept are that the hole will be drilled, or a shaft excavated, in a regime of moderate to low geothermal gradient in rock with high strength and low permeability. Furthermore, the wastes are to be deposited irretrievably - not stored (LBL 1979). The specific geotechnical considerations are addressed in detail in LBL (1979) and Brace (1979).

Since more holes would be needed, emplacement of spent fuel during a 40-year period would require a total land area of approximately 140 km². Canisters would be shipped by rail from a processing and encapsulation facility to the repository site, which would consist of a number of drilled holes around a centrally located receiving facility (Bechtel 1979a).

Waste Receiving Facility. The central waste receiving facility at the deep hole site would be used to unload the waste canisters, store them temporarily, and perform any work required to assure prompt emplacement in the hole. The receiving building would contain a cask handling area, a canister storage area, a hot cell, and auxiliary facilities (see Bechtel 1979a).

The cask handling area would contain facilities for receiving, cleaning, and decontaminating shipping casks and for reloading empty casks on rail cars. Upon arrival, an overhead

bridge crane would remove the loaded shipping cask and move it to the confinement section of the building. The lid would be removed and the cask aligned with a hot cell port. The HLW or spent fuel canisters would be removed remotely to a storage rack within the hot cell.

An interim dry storage area adjacent to the hot cell would have space for a 1-month supply of canisters.

The hot cell would include space for checking the canisters for visible damage, radiation leakage, and surface temperature. Facilities would be provided to decontaminate waste handling equipment in case of a canister failure. Damaged canisters would be overpacked and returned to the processing and emplacement facility for repacking.

The receiving facility would also provide auxiliary services such as ventilation, equipment maintenance, and a control system.

Canister Transporters. Canister transporters, similar to those used for subsurface transportation and emplacement in the mined geologic repository (Section 5.4), would be used to transfer the waste from the receiving facility to the emplacement facilities. Each transporter would consist of a wheeled vehicle suitable for operation on site roadways, a shielded transfer cask, and equipment for raising and lowering canisters in and out of the transfer cask. In the receiving facility, the transporters would be positioned over a portion of the hot cell to bottom load the canisters into the transfer cask. At the emplacement facility, the transporters would be positioned over the temporary storage area and the canisters would be bottom discharged into temporary storage.

Drilling System. The drilling rigs would be similar to those used in the gas and petroleum industries and would be portable for movement from one hole location to another on the site. Each complete rig would require a clear, relatively flat area, approximately 120 x 120 m (400 x 400 ft), at each hole location (McClellan 1977).

In the reference concept, the drilled hole for spent fuel is 48 cm (19 in.) in diameter and 10,000 m deep (Bechtel 1979a). For HLW, the hole is 40 cm (16 in.) in diameter. The depth and diameter, however, could vary depending on the geologic medium, the depth required to satisfy containment requirements, and the drill rig capacity. For HLW, the hole would be fully cased to the required depth with seamless steel pipe about 40 cm in outside diameter, which would reduce the hole diameter available for waste.

Oil field rotary drilling techniques would be used to sink the holes, which may be stepped down in diameter as the depth increases. To seal the pipe to the rock, a grout would be forced through the pipe and then back up between the wall of the hole and the outside of the casing. The bottom of the hole would be sealed.

During the drilling and emplacement operation, the hole would be kept full of drilling mud to facilitate drilling, prevent casing and canister corrosion, minimize casings sticking to the sides of the hole during installation, and counter lithostatic pressure.

Emplacement Facilities. Each emplacement facility would include a confinement enclosure to provide a controlled environment for emplacement operations, and the temporary canister

storage facility (Bechtel 1979a). The entire emplacement facility would be on rails for movement from hole to hole on the site.

As described above, canisters would be transferred from the receiving facility to the temporary storage facility, which would provide shielding and an accumulation area for canisters to accommodate differences between transfer and emplacement operations. Emplacement equipment with cable totaling at least 10,000 m in length would lift a waste canister from temporary storage into a shielded cask, position it over the very deep hole, and lower it through the bottom of the cask into the hole (Bechtel 1979a). The waste canisters would be lowered into the lower 1,500 m (5,000 ft) of the hole with metallic honeycomb spacers placed between each canister to absorb impact in case a canister is dropped (Bechtel 1979a). If required by canister structural design limits, a structural plug, anchored to the sides of the hole, would be emplaced between groups of canisters to support the load.

Sealing Systems. After all waste canisters are in place, the hole would be sealed to isolate the waste from the biosphere. Sealing could include plugging both the hole and the damaged rock zones around the hole.

The components of the sealing system would have to have low permeability to limit nuclide migration and sufficient strength to maintain mechanical integrity for a specified period. Possible plugging materials include inorganic cements, clays, and rock. The specific material or materials would be selected for compatibility with the geologic medium and down-hole conditions (Bechtel 1979a). Plugging could be done with standard equipment typically used by a drilling rig crew. For final sealing and closure of the very deep hole, drill rigs, similar to those described for hole drilling, would be set up at the hole location.

Retrievability/Recoverability. Waste canisters would be retrievable as long as they are attached to a cable during the emplacement process. Once the canister is disengaged, it would become essentially irretrievable. Post-enclosure recovery is likewise considered nearly impossible.

6.1.1.3 Status of Technical Development and R&D Needs

Present State of Development

The status of equipment facility, and process development for different operational phases of VDH emplacement are considered below.

Drilling Techniques. Four methods to excavate a very deep hole have been considered. These are oil field rotary drilling, big hole drilling techniques, drill and blast shaft sinking, and blind hole shaft boring. The latter three methods are limited in the depths that can be attained at present and in the foreseeable future. They might have applications in specific geologic media but will not be considered further here since the possibility of their use appears remote for waste emplacement in this concept. For details on these concepts, see LBL (1979).

For oil field rotary drilling, standard oil field drill equipment would be used. In this method, a drill bit attached to a drill pipe is rotated from the surface, and drilling mud is circulated through the drill pipe to carry cuttings to the surface. The drilling mud also assists in providing borehole stability, provides lubrication and cooling, and minimizes pipe sticking. Substantial rotary drilling experience exists; however, most of the drilling has been in sedimentary formations.

At least the upper portions of deep rotary drilled holes would be cased; and, in fact, the entire hole may need to be cased for borehole stability, as in the reference concept (LBL 1979). As described there, cement grouts would be pumped from the bottom of the hole up around the steel casing to seal the casing against the drilled borehole. If the entire borehole were cased, then the hole could be bailed dry (depending on the depth of the hole), and could be left standing open for extended periods. If the bottom portion of the hole were not cased, it is unlikely that the borehole would stay open if the hole were bailed dry. Some fluid, probably with a density somewhat higher than that of fresh water, would therefore be required in the open hole at all times.

There is little experience at drilling in hard, crystalline rocks, although such rocks may pose no more, or fewer, problems than drilling ultra-deep wells in sedimentary rocks. A limited number of oil field rigs are capable of drilling to 8,000-m (25,000 ft) depths and beyond, and there are presently four rigs in the U.S. capable of drilling to a depth of 9,000 m. The bottom portions of such holes have been drilled with a 16.5 cm (6-1/2 in.) diameter bit, and the holes were cased to the bottom. There is some experience in drilling geothermal wells where formation temperatures are 30-0 C (approximately 600 F) as anticipated in VDH drilling; however, these wells have not been drilled much below 3,000 m (10,000 ft).

It is believed that deeper and larger diameter holes could be drilled. A maximum well depth of about 11,000 m (36,000 ft) in rocks where borehole stability is not a problem is believed possible, using a 20-cm (7-7/8 in.)-diameter bit for the bottom hole. Depths of 9,000 m could be achieved with 31-cm (12-1/4 in.)-diameter bits in crystalline rocks where no gas pressure exists. For very strong rocks, the bottom part of the hole might be left open. In fact, for the 31-cm-diameter hole, the bottom part of the hole may have to remain open because current rigs (with current casing) would not be able to set casing to the bottom of a 9,000 m hole. A drill rig with a 15,000-m (50,000-ft)-depth capability has been designed but not operated which would utilize the largest available components. It would provide a 22-cm (8-1/2-in.)-diameter hole at total depth (Drilling DCW 1979). Salt has been drilled successfully to about 4,600 m (15,000 ft); below this, borehole closure prohibits further drilling.

Emplacement. The technology for emplacing waste canisters is not fully developed at present. Some technology for emplacing items to depths less than 10,000 m exists. For example, the Deep Sea Drilling Project has a hydraulically operated down-hole device that disconnects the boring bits.

Sealing. Standard oil field practices for cementing in casing have satisfactorily isolated deep high-pressure gas zones from shallower formations and from the surface for time periods measured in decades. Plugs of cement or other materials have been emplaced in abandoned oil and gas wells, both cased and uncased, and have maintained integrity over similar periods of time. In these instances, it is not uncommon for the casing to corrode prior to plug deterioration.

Logging/Instrumentation. Borehole geophysical logging techniques in existence and currently under development will permit the logging and analyses of a number of parameters critical to the emplacement of radioactive waste in very deep holes. Caliper, acoustic, televiwer, and other borehole geophysical devices are regularly used to verify the presence and distribution of fractures in well bores. Electrical logs, neutron porosity loss recorders, and other devices are used to verify the presence of water. Temperature logs and spinner logs are used to detect water flow. While all of this equipment can be used from depths of hundreds to thousands of feet, none of these tools can function at the temperatures [between 200 and 300 C (390 and 570 F)] and pressures anticipated at depths around 10,000 m, because of the electronics contained in the probe.

While rudimentary development of in situ stress measurements has been accomplished, the down-hole techniques would require significant improvement.

Issues and R&D Requirements

Depth of Hole. The hole depth required for adequate isolation from the biosphere would have to be determined by the geologic medium of interest and by the history and physical condition of that medium. Sedimentary rocks in some instances are considered as potential VDH locations, but only where they are considered to be lower in elevation than circulating groundwater, such as deep basins or hydrologically stable synclines. Crystalline rocks may be the best geologic medium for VDH disposal. Usable hole depth in crystalline rock would be influenced by the depth of ground-water circulation within that rock. Ground-water circulation in weathered granite near the surface in a humid environment will generally be significantly greater than in fresh granite in an arid to semiarid environment.

R&D is required to determine the depth required in various geologic media to minimize the possibility of significant circulation to ground-water systems. The top of the emplaced waste would still have to be significantly below possible contact with circulating ground water, and would have to be properly plugged and sealed against such contact.

Drilling. The discussion of the present state of development of drilling makes it clear that emplacement of nuclear waste in very deep holes is not possible at this time given that (1) the waste canisters will be 31 to 36 cm (12 to 14 in.) in diameter and (2) the depth required for isolation from the biosphere may be as great as 10,000 m. If it is assumed that these two criteria are valid for the conceptual system, then a number of problems related to drilling would have to be solved to attain emplacement in very deep holes. The key issue is whether it will be possible to develop the technology to drill to 10,000 m with a bottom hole

diameter of approximately 48 cm (19 in.) so that a 36-cm canister could be placed in a mud-filled, fully cased hole.

No increase in the present capability to rotary drill deep wells is expected by the year 2000 without some very significant effort to develop new technology. Currently, there is no industry demand to produce the technology advancement necessary. If sufficient resources were available to advance technology, a 9,000-m hole with a 48-cm (19 in.) diameter might be attainable by the year 2000. Most of the hole would be cased; however, in high strength rocks without gas pressure, the bottom part of the hole might be left uncased. Technology improvements required to reach this depth include:

- New drilling muds capable of operating at temperatures of 370 to 430 C (700 to 800 F)
- High-temperature drill bits, either roller cone or diamond
- New drill pipe, including improved designs and use of improved (high-temperature) steels
- Improved support equipment, such as high-temperature logging and surveying tools and fishing tools
- Improved casing materials (high-temperature steels) and joint design
- High-temperature cements and surface pumps for pumping these cements.

Waste Form and Package Integrity. Criteria currently being proposed for waste forms and packages require total containment within the package for the time period dominated by fission product decay (up to 1000 years). The development of materials to retain their integrity for this period of time at temperatures that would be reached when the ambient rock temperature is 200 to 300 C and under geochemical conditions that would be encountered would require significant effort.

Heat Transfer (Thermomechanical and Thermochemical Factors). Under a normal geothermal gradient of 20 to 30 C/km (60 to 90 F/mi) ambient temperatures in excess of 200 to 300 C (390 to 570 F) are expected at a depth of 10,000 m. The heat released by radioactive decay of the emplaced waste would further increase the temperature of the surrounding rock. The magnitude of this induced temperature increase would be determined by the thermal properties of the rocks and the power output of the waste.

Because of the very large height-to-diameter ratio of the column of radioactive waste, the heat flux from the waste would be mainly in the radial direction, as from an infinite cylinder. The temperature within the heat source itself would be very nearly uniform and would drop very abruptly at the ends. Therefore, from a purely thermal point of view, this geometry would be very favorable. It takes 200,000 years for heat from 5,000-m depths to diffuse to the surface (DOE 1979). The thermally induced effects on the chemical stability and mechanical integrity of the geological formation and upward driving of the ground water would be the most critical issues.

The thermochemical behavior of rocks around a deep hole is not predictable at present. Since controlling factors would be the jointing, fracturing, and fluid content of the rocks,

thermomechanical behavior would need to be studied in situ. Heater tests in a variety of rocks at design depths would probably be necessary to understand the complex response to local high temperature of rock that is water saturated, stressed, and fractured.

Some aspects of thermomechanical behavior of rocks can be studied in the laboratory, however. Since fractured rock is in question, and since characterization of natural fractures is at present impossible, these laboratory studies would involve large samples of rock containing one or more joints, obtained by special sampling techniques. The samples may have to be large (dimensions of several meters). This would require extension of present laboratory testing techniques to test at conditions simulating the in situ environment. The areas where study would be particularly needed include:

- Thermal cracking and other forms of degradation of rock
- Thermoelastic response of intact and jointed rock over a long time frame
- Changes in permeability caused by heating a rock mass
- Two-phase transport of fluids in fractured rock
- Hydraulic fracturing in thermally stressed rock
- Thermal conductivity of hot, saturated thermally stressed rock
- Stress corrosion due to heated ground water in thermally stressed rock.

Emplacement. Most people engaged in drilling for resource exploitation feel that, to prevent collapse, the borehole would need to be kept full of drilling mud at all times. This would include the period during which the canister would be lowered for the waste disposal concept. Getting the waste canister to drop through the drilling mud could be difficult because of the close clearance between the casing and canister. The potential accidental contamination of the drilling mud and lowering cable should a waste package be ruptured would raise numerous questions regarding decontamination techniques and optimum loading methods.

Thus, in addition to a need for substantial research and development on improving the properties of the drilling mud, techniques and equipment would have to be developed to assure lowering and releasing the canisters at depths of 10,000 m and for decontaminating the drilling mud and cable in case of canister failure during this operation.

Isolation from the Biosphere. The principal issue of radioactive waste emplacement in very deep holes is the long-term isolation of the waste from the accessible biosphere (LBL 1979).

In addition to packaging, hole conditions, and hole sealing, a number of other conditions would have to be addressed before long-term isolation from the biosphere could be assured. Several of these involve geotechnical considerations, including:

- An improved understanding of the hydrologic regimes of deep crystalline and sedimentary rock units, including porosity, permeability, and water presence.

- An improved understanding of in situ rock mechanical properties under the high temperature and pressure conditions expected at the required depths and under unusual thermal loading conditions. These properties include strength, deformation, stress state, and permeability.

Additional R&D might be required in the areas of site selection, site evaluation, and geochemistry (LBL 1979).

Sealing. It is assumed that the sealing system for very deep holes must meet the same time requirement for sealing penetrations used by mined repositories. The primary purpose of the seal would be to inhibit water transport of radionuclides from the waste to shallow ground water or the surface for the specified time period. For integrity to be maintained, the sealing material would have to meet the following requirements.

- Chemical composition - the material must not deteriorate with time or temperature
- Strength and stress-strain properties - the seal must be compatible with the surrounding material, either rock or casing
- Volumetric behavior - volume changes with changes in temperature must be compatible with the enclosing medium.

The seal system would consist not only of plugs within the casing, but also of material to bridge the gap between the casing and surrounding rock. To minimize the possibility of a break in containment, rigorous quality assurance would be required during the placing of several high-quality seals at strategic locations within the borehole.

Therefore, research and development would be needed in two major areas - materials development and emplacement methodology - to ensure permanent isolation. Materials development would include investigating plugging materials, including special cements, as well as compatible casing materials and drilling fluids, which might be incorporated into the sealing system. Because the seal would include the host rock, these investigations should include matching plug materials with the possible rock types. It is conceivable that different plug materials would be required at different points in the same hole.

Emplacement methodology would have to be developed for the particular environment of each hole. Considerations should include all envisioned operations in the expected environment, casing and/or drilling, and fluid removal. Because the emplacement methodology would depend on the type of sealing material, initial studies of sealing material development should precede emplacement methodology development. However, the two investigations would be closely related and there should be close interaction between the two phases. In situ tests should be performed to evaluate plugging materials. Equipment developed should include quality control and quality assurance instrumentation.

Logging/Instrumentation. Proper development and operation of a VDH emplacement system would require the collection of reproducible, remotely sensed data on the geologic formation from the bottom of a borehole under high temperature and pressure. Existing logging tools are generally not designed to operate at temperatures exceeding 175 C (350 F).

Remote determinations of water content and flow and in situ stress would need to be addressed to permit preplacement assessment of down-hole conditions to facilitate VDH system design.

Much of the R&D work under way for logging and instrumentation equipment would be applicable to monitoring equipment for the waste disposal area (DOE 1979).

R&D Costs/Implementation Time

The total cost for research and development for this concept is estimated to be about \$730 million (FY 1978 dollars) as derived from DOE (1979). The major portion of this cost, or about \$600 million, would be for development of drilling techniques and equipment. The development activity described could be accomplished over a 12 to 15-year period.

Summary

Major uncertainties, shortcomings, and advantages of the concept are summarized below:

- The capability to drill with diameters up to 50 cm holes to a depth of 10,000 meters does not exist and would require a tremendous advance in the state of technology. However, should it be demonstrated that considerably lesser depths, e.g., 3,000 m, are consistent with the concept they can be currently achieved with holes of adequate size.
- The temperature, pressure, and chemical environment at depth would present a potentially very hostile environment for the waste package. Significant advances in materials technology might be required to ensure long lived package design.
- Corrective action, defined as retrievability of emplaced waste, would be unlikely after emplacement.
- The approach is probably not consistent with the philosophy of being able to demonstrate technical conservatism in that design margins are considered small.
- Current methodology does not permit adequate assessment of the at-depth emplacement environment, nor are criteria available for site selection.
- The extreme depth of the concept, and the resulting lengthy path to the biosphere might compensate for many of the drawbacks.

6.1.1.4 Impacts of Construction and Operation (Preplacement)

During the construction and operation phases, the environmental impacts of the VDH concept would be those common to other drilling and excavation activities. Drilling the hole would raise environmental considerations similar to those for drilling deep holes for oil and gas wells, for uranium exploration and production, and for geothermal and deep rock mining. VDH impacts for these phases would be: the conversion each year of several square kilometers from present land uses to drilling/mining and waste repository activities; disturbance and removal of vegetation; temporary impoundment of water in mucking and settling ponds; accumulation of tailings; alteration of the topography at, and adjacent to, the site; and socio-economic impacts on housing, schools, and other community services. No special environmental considerations beyond those required for normal drilling would be required.

Health Impacts

Radiological Effects to Man and Environment. As indicated earlier, two different waste forms could be considered for disposal in very deep holes: spent fuel in canisters and encapsulated processed high-level waste. A detailed description of these forms is contained in Bechtel (1979a). Additional assumptions are that both waste forms would have undergone a 10-year decay period prior to emplacement and that secondary TRU wastes would be disposed via a mined geologic repository.

The estimated total occupational whole-body dose from VDH disposal during routine operations would be 4,150 man-rem/yr for the spent fuel waste form and 6,260 man-rem/yr for the HLW form (Table 6.1.1). Of this, 910 man-rem/yr for the spent fuel waste and 920 man-rem/yr for the HLW form can be attributed to the emplacement of waste in the deep hole. The detailed breakdown of doses directly attributable to the VDH concept is presented in Table 6.1.2. Doses attributable to the naturally occurring radioactive materials released during excavation of very deep holes are not included in the estimates.

The estimate of the total nonoccupational whole-body dose from VDH disposal is 380 man-rem/yr for the spent fuel waste form and 180 man-rem/yr for the HLW form (see Table 6.1.1.). Only a very small portion would be contributed by the deep hole -- 7×10^{-6} man-rem/yr and 3×10^{-4} man-rem/yr, respectively, for the spent fuel and HLW forms.

Only nonoccupational doses have been estimated for abnormal conditions and these are presented in Table 6.1.3. Insufficient data are available to allow an estimate of the exposure to occupational personnel during abnormal conditions. It can be only assumed that the exposure would be within regulatory requirements. In this instance, the estimated total

TABLE 6.1.1. Radiological Impact - Routine Operation (Bechtel 1979a)

	<u>Whole Body Dose, man-rem/yr</u>	
	<u>Occupational</u>	<u>Nonoccupational</u>
<u>Spent Fuel</u>		
AFR	1580	320
Packaging and Encapsulation (P/E) Facility	1100	20
Transportation	80	40
Repository (secondary waste)	470	5×10^{-6}
Deep Hole	920	7×10^{-6}
Total	4150	380
<u>HLW</u>		
P/E Facility	4090	90
Transportation	210	90
Repository (secondary waste)	1030	2×10^{-5}
Deep Hole	930	3×10^{-4}
Total	6260	180

TABLE 6.1.2. VDH Concept - Occupational Doses During Normal Operation (Bechtel 1979a)

Operation	Whole Body Dose, man-rem/yr	
	Spent Fuel	HLW
Primary Waste Receiving	170	220
Damaged Canister Receiving/Processing	80	100
Surface Waste Management	40	70
Decommissioning	40	10
Primary Waste Placement	370	320
Interim Confirm. Building	30	30
Support/Overhead	180	170
Total	910	920

whole-body dose would not be applicable because the individual estimates given in Table 6.1.3 cannot be added algebraically. However, note that for both waste forms the potential for the highest exposure would be for a transportation accident, which is not an operation unique to the VDH concept.

Nonradiological Impacts. Nonradiological impacts should be comparable to those of any large construction project and those of industry during operation. Injuries, illnesses, and deaths common to such operations might be expected.

TABLE 6.1.3. Radiological Impact - Abnormal Conditions(a)

Operation	Whole-Body Dose, m rem/event
	(Nonoccupational)
<u>Spent Fuel</u>	
AFR	2×10^{-3} (b)
P/E Facility	3×10^{-1}
Transportation	1100(c)
Repository (secondary waste)	60(d)
Deep Hole	60
<u>HLW</u>	
P/E Facility	3×10^{-1}
Transportation	1100(c)
Repository (secondary waste)	60(d)
Deep Hole	70

- (a) Dose estimates imply consequences of a design basis accident. No probability analysis is included.
 (b) Design base accident (DBA) is tornado.
 (c) DBA is train wreck, in urban area followed by a fire.
 (d) DBA is hoist failure handling secondary waste.

The occupational hazards during normal operations of the waste disposal system would be expected to be no more, and maybe fewer, than the average associated with the various trade/professional workers required to operate the system.

In the case of routine operation nonoccupational hazards, the expected impact would not be detectable.

There are no specific data available to permit a quantitative estimate of the consequences of accidents that may arise. It is expected that abnormal occurrences such as fires, derailments, transportation accidents, and equipment failures common to industry would occur, but with reduced frequency. Consequently, the occupational impact would be expected to be less than that for industry in general.

Natural System Impacts

Currently available information is so limited that quantitative estimates of the radiological impact on the ecosystem are not available. However, it is expected that, during normal operations, the impact would be minimal, i.e., not greater than that for the mined geologic repository concept. Engineered safety features would be provided to ensure that the disposal system would operate in compliance with regulatory requirements. In addition, location of the waste in holes as deep as 10,000 m would increase the transport path to several kilometers more than that for the mined geologic repository. This would tend to further mitigate the consequences of radioactive waste leak, should it occur, by increasing the transport time.

Microfractures and other openings might develop in the vicinity of the hole because of the stress relief created by drilling or excavation. In addition, small openings might develop within the cement plug and between the plug and the hole wall if the bonding between the two were not adequate. Such channels would provide pathways for contaminated waters to migrate to the biosphere. If the hole were sited below circulating ground water, the primary driving force for migration would likely come from the thermal energy released by the radioactive waste. The travel time to the biosphere would therefore depend on the availability of water, the continuity and apertures of the existing and induced fractures, the time and magnitude of the energy released, geochemical reactions, and the volume and the geometry at the opening over which the energy persists. The lack of data on the presence of water and the properties of fractures in deep rock environments prevents making any estimate of the consequences to the ecosystem.

Nonradiological effects on the ecosystem might impact both water and air quality. Water quality might be affected by the discharge of treated wastewater to the surface water and by rainfall runoff from graded areas, rock piles, and paved areas. Air quality and meteorological changes would come from the generation of fugitive dust and the creation of reflecting surfaces. Air quality would also be affected by emissions from diesel-powered construction and transportation equipment, stack gases, and fugitive dust. The exact discharge quantities and runoff characteristics and the exact amount and type of construction equipment are not

available at this time. Parameters such as vehicle miles, surface areas of structures and pavement, soil characteristics, and size of stock piles are also unavailable. For each of these parameters, a qualitative estimate was developed where the water quality effects are based on total land requirement for the facility. The meteorology and air quality impact estimate was based on the number of construction sites, which represent a variety of dust and diesel emissions, and the number of operational emission sources (Bechtel 1979a). The estimates are given in Table 6.1.4.

Socioeconomic Effects

A complete assessment of the socioeconomic impacts of the VDH concept cannot be made at this time because few data are available. In addition, the data that are available can be used only inferentially. These data, which relate to operating employees and community facilities, indicate that impacts would be only moderate.

These inferences are based on a classification scheme where minor, moderate, and major correspond to less than 2,000 employees, between 2,000 and 4,000 employees, and more than 4,000 employees, respectively. For the community facilities two locations is minor, three to ten locations is moderate, and more than ten locations is a major impact.

Aesthetic Effects

As with socioeconomic effects, only minimal data are available for aesthetic effects and these data can be used only inferentially. The available data relate to visual effects only. In this case, the inference is that aesthetic impact would be moderate for both waste forms.

This inference is based on a classification scheme where:

- Minor = no permanent structures, facilities, or equipment more than 100 m high
- Moderate = one facility with permanent structures, features, or equipment more than 100 m high
- Major = more than one facility with permanent structures, facilities, or equipment more than 100 m high.

TABLE 6.1.4. Nonradiological Environmental Impact

Category	Spent Fuel	HLW
Water Quality Facility Area, ha	2400	800
Meteorology and Air Quality, number of construc- tion sites/operational sources	9/42	0/10

Resource Consumption

The consumption of major resources for each case has been estimated from available literature.

Energy. The estimates of energy consumption in the forms of propane, diesel fuel, gasoline, and electricity are presented in Table 6.1.5 for both the spent fuel waste form and HLW (Bechtel 1979a).

Critical Material Other Than Fuel. The estimated consumption of critical resources is presented in Table 6.1.6 (Bechtel 1979a).

Land. The estimated total land that would be required for a 5,000 MTHM/yr waste disposal system is 14,000 ha (35,000 acres) for the spent fuel waste form and 8,000 ha (20,000 acres) for the HLW form. In both cases, the estimated impact would be moderate.

International and Domestic Legal and Institutional Considerations

The international/domestic legal and institutional considerations associated with a VDH repository are expected to be of the same nature as those addressed for a mined geologic repository. (See section 3.3.2 and section 3.5.2)

6.1.1.5 Potential Impacts Over the Long Term (Postemplacement)

The potential for impacts over the long term would relate both to human activities and to natural phenomena. In turn, human activities could be related to the failure of engineered features or human encroachment. Natural phenomena, such as earthquakes and volcanoes, could also degrade the integrity of the waste repository. The heating, rock alteration, or thermo-mechanical pulsing that could be caused by wastes reaching critical mass are issues common to other geologic disposal alternatives. These aspects would be dependent on the specific rock and site characteristics, waste form, quantity, and spacing and could be evaluated only when these parameters have been defined.

Table 6.1.5. Estimated Energy Consumption

Fuel Type	Spent Fuel	HLW
Propane, m ³	2.3 x 10 ⁴	1.0 x 10 ⁷
Diesel, m ³	1.6 x 10 ⁷	3.4 x 10 ⁶
Gasoline, m ³	1.6 x 10 ⁵	1.2 x 10 ⁵
Electricity, kWh	2.0 x 10 ¹⁰	5.6 x 10 ¹⁰

TABLE 6.1.6. Estimated Consumption of Critical Resources

Material	Spent Fuel	HLW
Carbon Steel, MT	3.3×10^6	6.8×10^5
Stainless Steel, MT	8.4×10^4	2.3×10^4
Components		
Chromium, MT	1.4×10^4	4.6×10^3
Nickel, MT	7.5×10^3	2.0×10^3
Tungsten, MT	3.0×10^3	0.5×10^3
Copper, MT	1.3×10^3	1.9×10^3
Lead, MT	1.3×10^4	2.9×10^3
Zinc, MT	1.2×10^3	0.6×10^3
Aluminum, MT	1.3×10^3	1.2×10^3
Water, m ³	2.0×10^8	5.9×10^7
Concrete, m ³	1.9×10^6	1.3×10^6
Lumber, 10 ⁴ m ³	5.6×10^4	3.8×10^4
Clays, 10 ⁶ MT	9.2×10^6	1.5×10^6

Potential Events

The long-term impact of a VDH repository on the ground-water regime would be governed essentially by the nature of the deep ground-water system. Because of the great depth of emplacement and the larger volume of rock available to absorb the energy released by radioactive decay, the deep ground-water system probably would not be appreciably perturbed by the waste itself. If the deep hole were located within a recharge zone or in a zone of lateral movement, the distance to the biosphere along the path of flow might be so long and the velocities so low that isolation might be effectively achieved. Furthermore, the transport of radioactive contaminants by the flowing water would also be greatly retarded by the increased residence times and the increased time for interaction of the contaminant with the host rock.

Engineering Failure of Isolation Mechanism. The principal engineered isolation mechanism for this waste disposal system would be the containment seal. After emplacing the nuclear waste in the deep boreholes, the holes would be sealed to isolate the waste from the biosphere. This isolation would have to be sustained for tens to hundreds of thousands of years for HLW. Not only would it be necessary to seal the borehole itself, but consideration would have to be given to plugging any damage that could have occurred around the hole.

The loss of the integrity of this containment seal might provide a pathway for the waste into the biosphere. The impact on the environment resulting from such a failure could be

evaluated only on the basis of site-specific parameters. The lack of specific data prevents a quantitative evaluation. However, it is not expected that resulting impacts would be any greater than those for a mined geologic repository under comparable conditions and might be less due to the longer pathway of smaller diameter than a mine shaft.

Natural Phenomena. Another concern for the VDH concept in the long term would be the susceptibility of the ground-water system to tectonic changes and volcanic action. The very concept of the deep hole is aimed at minimizing such effects by increasing the distance to the biosphere as much as is technically feasible. Placement of the waste disposal site in a tectonically stable region would reduce the probability of such catastrophic events. Site-specific data would be required to quantitatively assess the impact of natural phenomena leading to degradation of the containment.

Inadvertent Human Encroachment. Human intrusions into the VDH repository in the long term could result from drilling, exploration, and excavations. Monitoring, surveillance, and security operations carried out after the repository were closed would provide an increment of safety against such occurrence. However, the physical depth of the VDH would in itself be expected to provide a significant deterrent against human encroachment.

Potential Impacts

The loss of integrity of the waste disposal system as a result of an engineered system failure, natural phenomena, or human encroachment might give rise to environmental consequences by introducing radioactive waste into the biosphere, which would result in radiological health effects. Similarly, ecosystem effects and nonradiological health effects are conceivable.

Radiological Health Effects. It is difficult to predict the nature of future events that would cause a breach of the barriers isolating the nuclear waste from the biosphere. Hence, it is assumed that the system would perform as designed for a prespecified period of thousands of years (Bechtel 1979a). After the period in which the isolation scheme performs as engineered, the barriers would be assumed to be susceptible to breach by:

- Normal degradation, due to expected, naturally evolving events, such as breach by an aquifer with the eventual leaching and migration of the waste
- Abnormal penetration, due to unexpected events, such as drilling or mining of the waste site by man.

The actual scenarios are described in detail in Bechtel (1979a). The radiological impact is expressed in terms of dose per year or dose per event in the case of the abnormal occurrence. The impacts are given in Table 6.1.7.

Ecosystem Effects. An evaluation of the effects on the ecosystem in the long term requires data that are presently unavailable. However, it is not expected that the impact on the ecosystem would be any greater than that for a mined geologic repository, and maybe less, since the radionuclides would be expected to take longer to reach the biosphere.

TABLE 6.1.7. Long-Term Radiological Impact of Primary Waste Barrier Breach

	Waste Type	
	Spent Fuel	HLW
Normal Events (mrem/yr)		
Whole Body	7×10^{-4}	7×10^{-4}
Bone	5×10^{-4}	5×10^{-4}
Abnormal Events (mrem/event)(a)		
Whole Body	Negligible	Negligible
Bone	Negligible	Negligible

(a) Dose is 50-year dose commitment from 1 year intake to the maximum exposed individual.

Nonradiological Health Effects. Although there are no specific data to evaluate the non-radiological health impact, it is expected that these impacts would be comparable to those found in the corresponding industries, e.g., mining, drilling, and excavating.

6.1.1.6 Cost Analysis

All cost estimates are in 1978 dollars based on January 1979 dollar estimates (Bechtel 1979a) less 10 percent.

The estimates are based on preliminary conceptual design data and were developed without the aid of previous cost estimates for this type of facility. Because of the high uncertainties in the cost of rotary drilled holes as large and deep as are called for in this VDH concept, the costs given should be considered only as preliminary estimates.

Capital Costs

On the basis of the waste system description, as presented in Section 6.1.1.2, the estimate of the capital cost for the spent fuel case is approximately \$2.3 billion. For the HLW case, a capital cost estimate is \$290 million (Bechtel 1979a).

Operating Costs

Operating cost estimates for the spent fuel case have been calculated per year for years 1 through 38 and then for phasedown years 39 and 40. These costs, which include VDH rotary drilling, moving emplacement structures, hole sealing, and receiving facilities operations, would be about \$1.7 billion for each year through the 38th year, \$1.6 billion for year 39, and \$0.8 billion for year 40.

For the HLW case for the same time periods, estimated costs would be \$210 million for each year through the 38th year, \$200 million for year 39, and \$260 million for year 40.

Decommissioning Costs

Total estimated decommissioning cost for the spent fuel case would be \$32 million. Total for the HLW case is estimated at \$11 million.

6.1.1.7 Safeguards

As noted, the waste types that can be handled in the VDH concept would be limited by volume constraints. Thus, choosing this alternative would require safeguarding two separate disposal flowpaths. The risk of diversion would be strictly a short-term concern, because once the waste had been successfully disposed of in accordance with design, the waste would be considered irretrievable. Physical protection of the sensitive facilities and transportation operations would be the most effective way to deny access to the waste for the short term, as is common to most waste disposal alternatives. For additional discussions of pre-disposal operations safeguards see Section 4.10.

6.1.2 Rock Melt

6.1.2.1 Concept Summary

The rock melt concept for radioactive waste disposal calls for the direct emplacement of reprocessed liquid or slurry HLW and remote-handled (RH) TRU into underground cavities. After the water has evaporated, the heat from radioactive decay would melt the surrounding rock, eventually dissolving the waste. In time, the waste-rock solution would refreeze, trapping the radioactive material in a relatively insoluble matrix deep underground. The waste and rock should achieve reasonable homogeneity before cooling, with resolidification completed after about 1,000 years. Rock melting should provide high-integrity containment for the radionuclides with half lives longer than this period. Spent fuel and secondary wastes (hulls, end fittings, and contact-handled (CH) TRU are not suitable for rock melt disposal unless they could be safely and economically put into a slurry for injection. Otherwise, they would be disposed of using some other form of terrestrial disposal, such as a mined geologic repository.

The waste-rock solidified conglomerate that would ultimately result is expected to be extremely leach resistant, to the extent that it might provide greater long-term containment for the waste isotopes than a mined geologic repository. Because less mining activity would be involved, the cost advantages could be substantial (Bechtel 1979a).

After emplacement, the waste would be considered to be irretrievable, although it could probably be recovered at great expense during the charging or waste addition period while cooling water was still being added. However, the recovery operation would become much more complex and expensive with time as the size of the charge increased (Bechtel 1979a).

There are several technological issues to be resolved and considerable R&D work would be needed before this concept could be implemented. Primary needs would be for better understanding of heat-transfer and phase-change phenomena in rock to establish the stability of the molten matrix and for development of engineering methods for emplacement.

6.1.2.2 System and Facility Description

System Options

The reference concept for rock melt disposal of nuclear waste has been developed from a number of options available at each step from the removal of spent fuel from the reactor to disposal in the rock melting repository.

Various options to be considered are summarized in Figure 6.1.3. The bases for selection of options for the reference concept (those blocked off) are discussed in detail in various documents listed in Appendix M. In addition, a number of options for variations within the concept were considered. These options could improve the concept by changing the cavity construction method or the waste form, or by eliminating cavity cooling (Bechtel 1979a and DOE 1979).

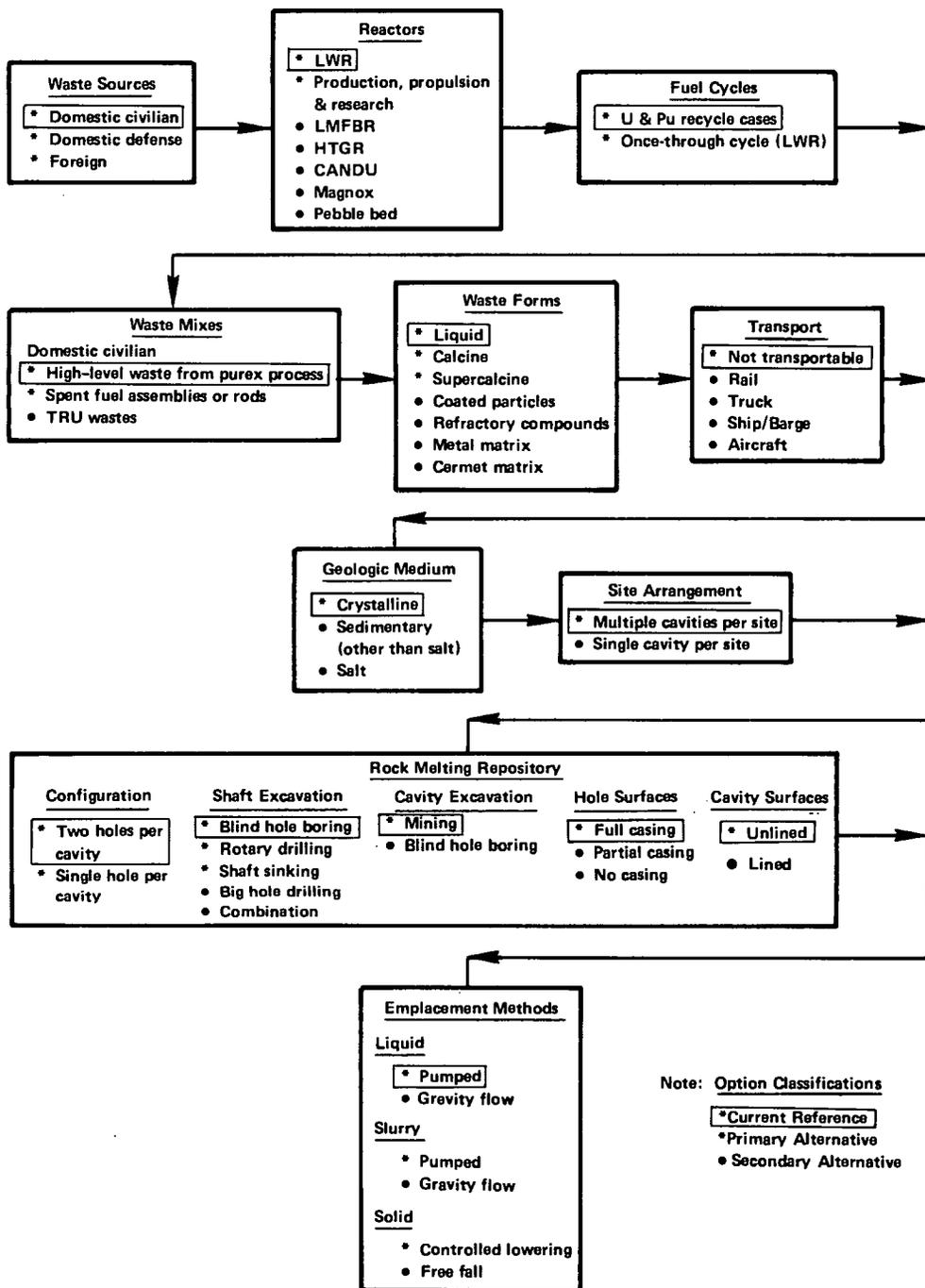


FIGURE 6.1.3. Major Options for Rock Melting Disposal of Nuclear Waste

Waste-Type Compatibility

It is assumed for the reference case that only liquid HLW and liquid RH-TRU would be injected into the rock melting cavity. Because of uncertainties associated with emplacement, such as additional criticality concerns, and a sufficient heat generation rate for the volume, spent fuel is not considered suitable for this reference case. Therefore, spent fuel and other wastes that may have low heat generation per unit of volume, such as solid RH-TRU and CH-TRU, are assumed to be sent to a geologic repository. Note that the suitability of spent fuel and other wastes for rock melt disposal may be improved by safely and economically putting them into a slurry form.

Waste-System Description

Basically, rock melting would work in the following manner. In the charging phase, HLW in aqueous solution would be injected into a mined cavity. The heat generated by the radioactive decay of the waste would drive off steam, which would be piped to the surface. When the boil-off rate reached a certain level, liquid transuranic wastes would be added to the charge. Periodically, high-pressure cleaning water would be flushed through the injection piping to minimize contamination and solid particle buildup. This cleaning water would also flow into the waste, providing a coolant to prevent the rock from melting during the waste charging phase. Cooling would be by evaporation or the heat of vaporization. At the surface, the steam driven off from the waste would be condensed and recirculated to cool the charge in the cavity. The closed system would be designed to prevent the release of radioactivity to the environment (Bechtel 1979a).

After about 25 years, when a substantial fraction of the cavity volume was filled, charging would be stopped. After the water was allowed to boil off and the waste to dry, the inlet hole would be sealed. The cavity temperature would rise rapidly and rock melting would begin, with radioactive materials dissolving in the molten rock. As the mass of molten rock grew, its surface area would expand and the rate of conductive heat loss to the surrounding rock would increase. Preliminary calculations indicate that at about 65 years, the rate of conductive heat loss from the melt pool would exceed the rate of heat input from radioactive decay. At this point, the melt would begin to slowly solidify. During the rock melting phase, the heat from the melt would inhibit ground water from entering the area and should prevent the leaching of the radionuclides. This is referred to as the "heat barrier" effect (DOE 1979). Following resolidification, when the heat barrier had dissipated, fission products would have decayed to very low levels. The relative toxicity of the residual radionuclides in the solidified waste-rock matrix is expected to be significantly less on a volumetric basis than that of a typical uranium ore from which nuclear fuel was originally extracted. The final product of the melt is expected to be a relatively insoluble sphere or resolidified silicate rock conglomerate, with a highly leach-resistant matrix, which would be deeply isolated from the biosphere (Bechtel 1979a).

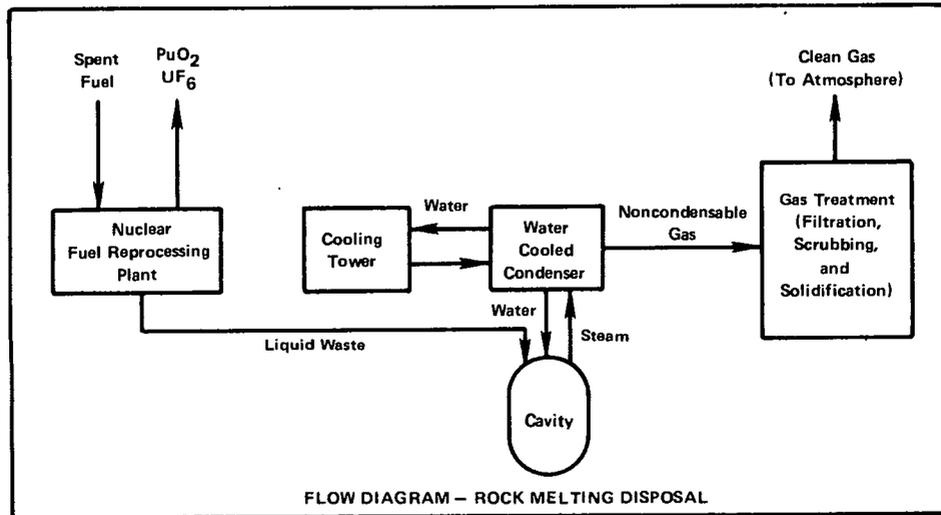
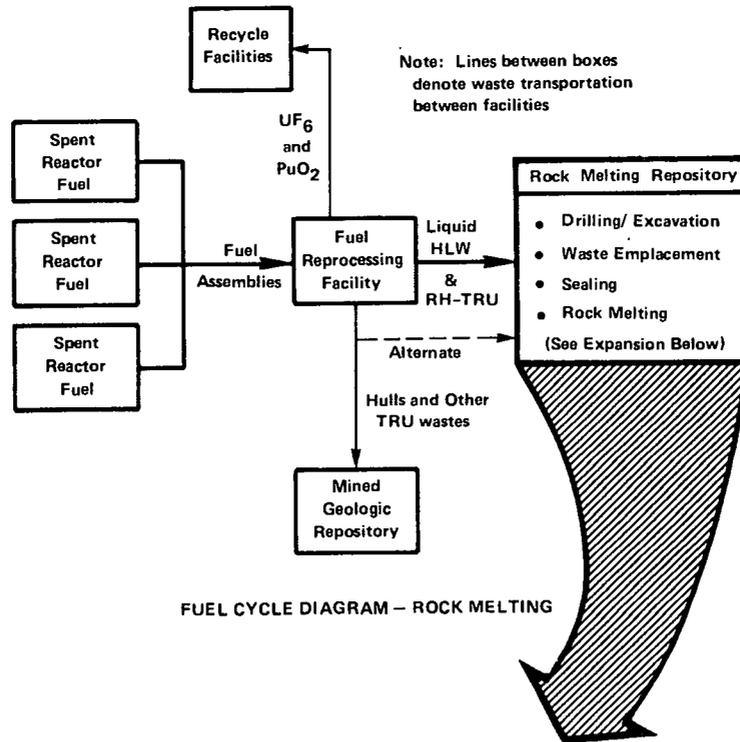


FIGURE 6.1.4. Waste Management System-Rock Melting Disposal

The reference concept design for rock melt disposal was selected through judgment of a "most likely" approach based on available information and data and is not supported by a detailed systems engineering analysis. The fuel cycle and process flow for this concept are shown in Figure 6.1.4. In the reference concept, a repository is designed for disposal of 4 million liters per yr (5,000 MTHM/yr) of high-level liquid waste (HLLW) for 25 years. This requires three 6,000 m³ (212,000 ft³) cavities, about 2,000 m (6,560 ft) below the surface on a single site. The three cavities would be located about 2,000 m from each other (Bechtel 1979a).

Predisposal Treatment of the Waste. The reference concept requires a fuel reprocessing plant to recover uranium and plutonium for recycle and to generate HLLW for disposal in the rock melting cavity, as described in Appendix VII of Bechtel (1979a). This plant could be located either on or off site, but the reference concept assumes an on-site location because of restrictions on the transportation of liquid radioactive materials. If solid pellets were produced in the packaging/encapsulation (P/E) facility, an off-site location would be feasible.

Site. The primary factor in selecting a site would be the suitability of the rock formations. Those rocks of greatest interest as potential media for rock melt disposal are composed of silicate minerals. Silicate mixtures are characterized by a melting interval rather than a definite melting point, the melting interval being different for each different set of minerals (DOE 1979).

The melting interval is bounded by the solidus temperature (the temperature at which liquid first forms as the rock is heated) and the liquidus temperature (the temperature above which mineral crystals do not exist stably). In rock melting, these temperatures would depend on parameters such as pressure, chemical composition (especially the amount of water present) and the state of segregation of the rock (see Figure 6.1.5) (Piwinski 1967, Luth et al. 1964, and Wyllie 1971a). Therefore, the ultimate size of the rock melt cavity would depend on the waste decay heat level and the rock characteristics, including thermal conductivity and thermal diffusivity. Also, the ultimate volume of the molten rock would be influenced by the size of the original mined cavity. The radius of the waste-rock melt pool, as a function of time, for a typical rock melt repository is shown in Figure 6.1.6 (DOE 1979).

The total site area that would be required for a rock melt repository would depend on the number of cavities, the size of the cavities, spacing between the cavities, and surface facility requirements. For this reference concept, the site area would be approximately 4 km² (1.5 mi²) (Bechtel 1979a).

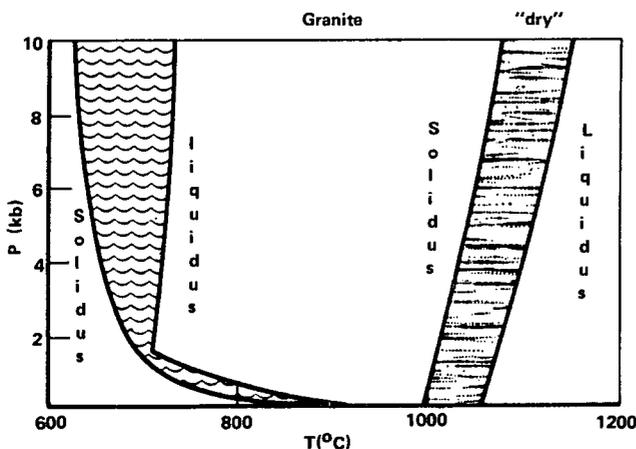


FIGURE 6.1.5. Schematic Illustration of Hydrous and Anhydrous Melting Intervals for an Average Granite

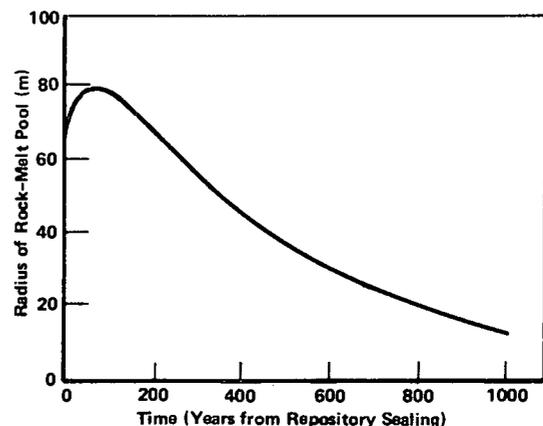


FIGURE 6.1.6. Radius of Waste-Rock Melt Pool Over Time (For Typical Cavity and Waste Loading)

Drilling/Mining System. The reference concept requires two access shafts for each cavity, each 2 m (6.6 ft) in diameter and approximately 2,000 m (6,560 ft) deep. They would be drilled using the blind hole boring method (Cohen et al. 1972). A rotating head with cutters would be turned by electric motors down hole. The entire boring machine would be held fixed in the hole by a hydraulic gripping arrangement. The shafts would be lined with carbon steel casings after drilling (Bechtel 1979a). This method would require men in the shaft to operate the boring machine (DOE 1979).

The cavity would be excavated by conventional mining techniques, although the equipment used would be limited by the access shaft diameter (Bechtel 1979a). Any blasting would be controlled to minimize fracturing of the surrounding rock. The spoil from both drilling and excavating would be hoisted up the access shafts by cable lift for surface disposal (Bechtel 1979a).

Repository Facilities. If the reprocessing plant were located on site, the reprocessing facilities would include a processing/packaging facility. If processing and packaging of wastes for off-site disposal were performed off site, the repository facilities would include a receiving facility similar to that described for the very deep hole concept (Section 6.1.1.1). The following description assumes that the reprocessing facility would be on site.

Four identical stainless steel tanks would be provided for storing HLLW. These tanks would have a combined capacity of about 10^6 liters (2.8×10^5 gal), which equals 3 months' production. The tanks, with the same design as those at the commercial reprocessing plant in Barnwell, South Carolina, would be contained in underground concrete vaults and provided with internal cooling coils and heat exchangers to prevent the waste from boiling (Bechtel 1979a).

An underground pipe system would connect the reprocessing facility to the storage tanks and the three rock melting cavities. The pipe would be double cased and protected by a concrete shielding tunnel. The pipe annulus would contain leak detectors. Heavy concrete and steel confinement buildings over the pipe and cavity shafts would provide for containment, shielding, monitoring, decontamination, maintenance, and decommissioning activities, primarily by remote control (Bechtel 1979a).

There would be four main pipes in the operating shaft to the rock melting cavity:

- A double-wall, stainless steel waste-addition pipe
- A single-wall, stainless steel water-cooling pipe
- A single-wall, stainless steel steam-return pipe
- A stainless steel instrumentation pipe through which monitoring devices would be inserted to measure the temperatures and pressures at various points in the system (Bechtel 1979a).

The confinement buildings over the cavities would also house the equipment and systems needed for filling the cavity and sealing the shaft. Three important process systems would

be: (1) the pipe and valve manifold enclosure, (2) the condensing plant, and (3) gas processing equipment. Pipe and valve manifolding would be located in an enclosure near the top of the cavity operating shaft. The cooling water injected into the cavity and the steam from the cavity would be routed through this enclosure. There would be an operating and instrumentation gallery adjacent to the enclosure (Bechtel 1979a). (The HLLW would be charged through a separate underground pipe, mentioned above, that would not go through the confinement building or the pipe and valve manifold enclosure.)

The condensing plant would cool and condense the steam coming out of the cavity and recycle it as cooling water during the waste charging phase. The potentially radioactive primary cooling loop and the nonradioactive, closed-circuit intermediate cooling loop, along with the associated pumps and heat exchangers, would be shop fabricated in modules and designed for rapid remote maintenance. Since the rock would start to melt in a matter of days without cooling, all heat exchanger and pump systems would be designed and constructed with full redundant capacity to ensure constant cooling.

Most of the gaseous elements in spent fuel would be removed during reprocessing at the fuel reprocessing facility. However, some fission product iodine in the liquid wastes could become volatile during the waste charging phase and would be carried out with the steam. This would be trapped by the gas processing equipment and returned with the cooling water to the waste charge or packaged for disposal in a mined geologic repository (Bechtel 1979a).

Auxiliary facilities would support the systems and equipment located inside the confinement building. These would include the water treatment plant, cooling tower, and radwaste treatment (Bechtel 1979a).

Sealing Systems. There would be two principal shaft sealing operations:

1. Sealing of the spare shaft after construction and before waste charging begins
2. Sealing of the charging shaft after completion of waste filling but before rock melting begins.

The NRC's Information Base for Waste Repository Design (NRC 1979) provides recommendations for sealing conventional boreholes and shafts. Though this information base may not be particularly applicable to the rock melt concept, it states that removal of the steel casing is essential for long-term performance of the seal. The seal must be bonded directly to the geological strata for maximum strength. Expansive concretes make the best seals under current technology and do so at an acceptable cost. However, it is not certain that these seals, whether cement, chemical, or other material, will successfully resist deterioration over a period of 1,000 years on the basis of current penetration sealing technology. Seal failure must be assumed even for seals placed under carefully controlled conditions using state-of-the-art technology and materials. Further development of sealing technology would, therefore, be required (DOE 1979).

Postplacement sealing of the pipes within the shaft, the shaft itself, and the pipes and valve gallery in the confinement building would be a more complex problem. This is be-

cause of the limited time, the high temperatures involved, and the radioactivity levels in the system. Considerable technology in this area has yet to be developed, as discussed in the following section.

Retrievability/Recoverability. Wastes disposed of by this concept would possibly be retrievable for a short period. Prior to melting, most of the liquid or slurry could be removed. After the melt has begun, well techniques for the molten rock-waste mixture might be possible. However this is unproven and would likely be an expensive and difficult process. Postclosure recovery of the solidified waste form would require extensive mining and excavation of large quantities of hot and molten rock containing waste.

6.1.2.3 Status of Technical Development and R&D Needs

Present State of Development

Substantial fundamental and applied research would be required for continued development of the rock melting disposal concept. This method is in the conceptual stage and no experimental work has been undertaken to support its feasibility.

Rock Melting Process. Generally, rocks are multiphase mixtures of a number of minerals characterized by a melting interval, as noted earlier. Because any two samples of a particular type of rock will have slightly different mineral compositions, they will also have slightly different melting intervals. As we have seen, the boundaries of these intervals (liquidus and solidus temperatures) depend on several parameters.

If the composition of the rock in which a waste repository were to be located has been well characterized, the melting properties of that rock could be predicted with some precision, and if the thermal conductivity, thermal diffusivity, and the heat of fusion of the rock were also known, the melting "history" of the HLW/rock melting phase could be predicted.

Clearly, it would be prudent to experimentally verify such predictions by means of prototype experiments; however, it should not be necessary to carry out an extensive series of such experiments to verify the current predictive capability for estimating the rate of rock melting and the total amount of rock melted for a particular set of waste repository conditions.

Effects of Heat on Rock Properties. The properties of rock subjected to high thermal gradients would be important inputs to determining the condition of the rock enclosing the molten waste-rock matrix. While the radius of this molten zone should be small compared with the extent of the geologic formation in which the repository would be sited, the zone's properties would have to be known so that an appropriate structural and safety analyses could be carried out.

The inner edge of this zone would be defined by the maximum radius of rock that had been heated to liquid formation. The outer radius of the zone could be roughly characterized as that location beyond which the rock had not been measurably affected by heat from the HLW.

The heat effects in the peripheral edges of the zone would be similar to effects found in a mined repository.

Transport of Radionuclides in Rock Melting. Under normal operating conditions, the casing in the emplacement well should prevent contact of radioactive waste with any aquifers that would overlie the disposal cavity. However, during waste charging, it is conceivable that some radioactivity could migrate out of the cavity into the surrounding rock. But, if the cavity were maintained approximately at atmospheric pressure, the tendency of water under hydrostatic pressure to flow into the cavity should minimize the importance of this transport mechanism.

During the rock melting phase, transport of radionuclides out of the waste-rock mixture would presumably be inhibited, because no water would be present in the melt and a portion of the surrounding zone of heated rock (Taylor 1977). (This is the "heat barrier" effect referred to earlier.) However, the radionuclide leaching capabilities of the high-pressure and high-temperature water vapor existing in this region would have to be characterized.

Finally, after the waste-rock matrix had cooled and solidified, it must be assumed that water would reenter the matrix and leach at least some of the radionuclides out of the matrix volume. Leaching potential at elevated pressure and temperature would have to be determined. As the radionuclides were transported to the relatively cool rock away from the repository, existing data on radionuclide transport in rock should be applicable (Klett 1974, Burkholder et al. 1977, de Marsily et al. 1977, Pines 1978, EPA 1978). It is possible that leaching data on other waste forms could also be useful (Brownell et al. 1974, Ralkova and Saidl 1967, Schneider 1971b, Mendel and McElroy 1972, Lynch 1975, and Bell 1971).

Effect of Superheated Water on Glasses in Rock Melting. Data from recent investigations of the devitrification of glass by water at high pressure and temperature (McCarthy et al. 1978 and McCarthy 1977) could be useful in determining the availability of radionuclides to water from vitrified rock present in the resolidified waste-rock matrix. However, the applicability of the conditions under which these data were obtained to the rock melt concept would have to be established.

Safety Studies: Disposal of HLW with Rock Melting. During the cavity charging portion of the presealing phase, HLW in such forms as solutions or slurries would be directly introduced into the repository cavity. The various operations that would be involved in carrying out this phase of the process are not as unique as the postsealing phase. Consequently, the probabilities for the release of radioactivity to the environment can be estimated for each step of this phase. This can be done both for normal operation and for assorted accident scenarios. In general, sufficient data exist to prepare a risk analysis for this phase of the rock melt concept.

After cooling of the waste-rock matrix to the point where water could contact the waste, it may be assumed for purposes of modeling that the waste dissolves, and transport through the surrounding rock is initiated. Calculations for risk analysis of this postsealing phase

are identical with those used for the risk analysis of other geologic waste disposal concepts with the exception of possible bulk migration of the molten mass during the interim phase between cavity sealing and solidification.

Ground Water Migration and Rock Melting. While a molten or high-temperature rock mass would disrupt natural patterns of water movement in the vicinity of a repository, the relative effect would diminish with distance, until, at some point, the repository would have no appreciable effect on water transport of radioactive materials. Presumably, if the hydrology of the repository area were well characterized, its effects could be modeled by treating it as a roughly spherical barrier with a radius that shrinks as the waste-rock matrix cools. Preliminary work on a laboratory scale and at atmospheric pressure indicates that this "thermal barrier" effect (Taylor 1977) could be demonstrated experimentally; however, additional work that more closely simulates conditions expected at the repository depth would be required.

Technological Issues

The technological issues that would require resolution before initiation of the rock melting concept can be summarized as follows:

- The necessary geological information cannot be predicted with present knowledge.
- Empirical data on the waste/rock interaction and characteristics are lacking.
- No technical or engineering work design of the required facilities has been attempted.

It is not possible at this time to produce a design for the rock melt repository because the necessary information is lacking. Data on the form and properties of the waste to be charged into the cavity, the charging methodology, the properties of the host rock, and many technical aspects of the shaft sinking method and cavity construction technique would have to be resolved. For many of these operations, work could not begin until fundamental waste/rock properties are better known.

In addition, the concept would require operations and process activities that do not readily lend themselves to the same degree of conservatism normally utilized in the nuclear field. Discussed below are several areas that would require further scientific or technical work.

Cavity Design and Construction. The greatest problem might lie in the construction of the cavity. Although, it is within the bounds of current technology to lower men and equipment through a 2-m-diameter shaft and construct the required cavity, such operations are difficult and time consuming. Methods for lining the cavity may have to be developed. Furthermore, it is practically impossible to construct the cavity without cracking the surrounding rock. Since it may be necessary to maintain the waste inside the cavity for some years before rock melting is permitted to begin, it would be necessary to ensure that waste does not escape into the cracks and ultimately into ground water. It may be difficult to assure

the necessary leaktightness of the mined out cavity. All of these areas would require technical resolution before construction could begin.

Cavity Charging. Cavity charging methods would depend on many variables including: the radioactivity of the charge; whether the charge were liquid or slurry; whether charging were batch or continuous; and whether charging were a long-term or short-term operation. The methodology for charging has not been defined or optimized. Considering the heat of the waste, the depth of the cavity, and possible corrosion and material plate-out, considerable technical effort would be required in this area.

In addition, the effect of a 2,000-m-long steam line on cavity charging would have to be determined. A vertical pipe of this length would act as a distillation column. Also, the engineering required to construct such a pipe (i.e., the number and type of expansion joints, effect of bends, etc.) has not been performed.

Shaft Sealing. There would be two phases of shaft sealing: sealing after construction but before waste charging starts and sealing after the waste is emplaced but before rock melting begins.

Sealing after construction would be the easier of the two operations because there would be sufficient time to check the work. However, sealing before rock melting begins would have to be done fairly quickly and in a potentially contaminated environment. Radioactive contamination and possible residual steam venting would present substantial problems in trying to seal the shaft after charging. Because of the number of pipes connecting the cavity to the surface, this operation would require considerable expertise. Both the materials and methods required would need further study and experimentation.

Volatile Fission Products. The quantities and behavior of the potentially volatile fission products would have to be determined. Nuclides in this category include ^{103}Ru and ^{106}Ru . Equipment would have to be designed to trap and remove these products from the waste stream or to return them in the coolant back to the cavity. Alternatively, they might be returned to the processing facility. There might also be a liquid and solid carryover from the steam, which would contaminate the condenser as well as increase the hazard from any potential leak. Practical technical considerations in this area would have to be examined before this concept could ever be considered viable. There is also a potential problem with tritium being carried with the steam.

Criticality Potential. Because 99.5 percent of the uranium and plutonium would have been separated from the spent fuel during reprocessing, the potential for criticality in the HLW is small. If experimental and modeling results indicated that criticality might be attained at some point in one of the rock melt concept scenarios, and if the results of such an excursion were undesirable from either an engineering or a safety standpoint, additional work would have to be carried out to develop methods of mitigation, possibly involving the addition of a high neutron cross section "poison" to the HLW as it is emplaced in the repository. It would be necessary for the "poison" to remain dispersed in the proper place upon cooling.

Fracturing During Cooling. During melting, the waste-rock mass would be expected to expand about 13 percent. During subsequent cooling and contraction, fracturing would have to be expected in the rock zone that surrounds the molten area. Further work would be required to establish that the rock melting concept could provide containment of the waste charge under uplift and subsidence conditions.

Chemical and Physical Effects on Surrounding Rock During Rock Melting. While the rock melting process can be described with some precision (Piwinski 1967, Luth et al. 1964, Wyllie 1971a, and Wyllie 1971b), the effect of a large thermal gradient on various types of rock has apparently not been similarly investigated (Executive Office of the President 1978). Although in some rocks, the predicted thermal effects of a molten mass of HLW/rock extend over relatively short distances, the extreme thermal gradient would clearly produce chemical and physical effects in the rock (Jenks 1977, National Academy of Sciences 1978). These effects would have to be characterized so that the rock mechanics of rock melt disposal could be adequately modeled and any possible intermediate or long-range effects identified and characterized. It would be necessary to carry out measurements over a range of pressures up to the maximum contemplated lithostatic pressure for a waste disposal cavity.

Interaction of HLW with Rock. At the present time, it is not clear whether the possible chemical reactions between the HLW solution and the rock cavity walls are important to the rock melt concept. However, it is clearly desirable to know how and to what extent such reactions take place, and to predict what the ultimate effect of 25 years of waste solution addition would be. With that information, potential problems could be identified, and mitigating measures could be designed and tested.

After addition of HLW to the cavity were stopped and rock melting begun, it is not known how rapidly and completely the HLW would mix with the molten rock. Because relatively complete mixing of the HLW with the rock appears desirable (to ensure complete dissolution of the HLW in the rock and subsequent immobilization upon resolidification of the matrix), it might be necessary to design the HLW rock melt disposal facility to minimize the viscosity of the molten rock.

Properties of Resolidified Waste-Rock Matrix. Even if it is assumed that the HLW is completely mixed with the molten rock, it is not known whether some of the radioactive species in the HLW might segregate during the long cooling process to form relatively concentrated (and possibly, relatively soluble) inclusions in the resolidified waste-rock matrix (Hess 1960). It is possible that the addition of certain chemicals (at the time that HLW is emplaced) could prevent such segregation, decrease the solubility of some or all of the long-lived radionuclides, or both.

R&D Requirements

Resolving these many uncertainties would require an extensive R&D program, such as that described below.

Data Base Development. Development of an adequate data base would require the conceptual design of one or more rock melt repositories. From these design bases, significant engineering features and critical geologic parameters could be identified. Similarly, the relevant properties of the geologic media would have to be understood in the context of the rock melt concept. Also, properties of materials in the waste handling systems would have to be identified and evaluated to determine the ability of these materials to function in hostile environments.

Laboratory-Scale Studies. To develop an understanding of rock melt mechanisms, extensive scale studies would need to be conducted. Specific areas of study should include:

- Heat transfer and phase-change phenomena for various geologic media
- Waste/rock interactions, particularly at elevated temperatures
- Properties of the resolidified waste-rock matrix
- Properties of engineering materials and their ability to function in the predicted environments
- Studies of actual small scale rock melt systems in laboratory hot cells
- Studies on the potential effects of criticality accidents.

Model Development. Better understanding of rock melt interactions could be gained by applying the data base to development of a predictive model covering heat transfer and related phenomena. The model could then be used for sensitivity analyses to determine the relative importance of various parameters and where research and development effort might best be applied.

Site Selection Methodology. From the systems modeling and other research tasks, it would be possible to identify those technological factors that would have to be considered in site selection. When site selection factors had been identified and evaluated, an optimal site profile could be determined to guide the selection process. Currently there is no methodology for locating a site.

Instrument Monitoring Techniques. Instrumentation for monitoring site selection and operational and postoperational phases of rock melt disposal would have to be identified and techniques for its use developed.

Thermal Analysis and Rock Mechanics. The effects of the melting cycle on the integrity of geologic formations would need to be thoroughly studied. Such effects as thermal expansion and contraction, phase change, and hydrologic change before and after emplacement would have to be assessed.

Pilot-Plant Studies. Laboratory and modeling studies should be complemented by a small-scale pilot-plant study involving actual emplacement of nuclear waste in rock. Such a study would be necessary to validate predictive methods and to assure that no vital factors had been overlooked prior to full-scale implementation of the concept.

Implementation Time and Estimated R&D Costs

In view of the significant technical uncertainties remaining, it is not possible to predict a cost estimate of the required R&D to implement this concept, nor the amount of time it would take.

Summary

Major uncertainties, shortcomings, and advantages of the concept are summarized below:

- There is not a multiplicity of engineered barriers inherent to the concept.
- The temperature, chemistry, and other characteristics of the molten waste-rock mixture are not considered consistent with technical conservatism.
- The required characteristics of a site are not known, and criteria for selection are considered extremely difficult to derive.
- The concept cannot be implemented in a step-wise, technically conservative manner due to the scale required for demonstration.
- Performance assessment capability is perhaps most distant for this concept than for any other.
- Retrievability of the waste is considered to be unlikely, so that corrective action cannot be accomplished.
- The time required for monitoring prior to full solidification (defined as the operational period of up to 1,000 years for this concept) exceeds the likely acceptable life for institutional controls.
- The primary postulated advantage relates to the possibility that the solidified waste form might be more stable than other possible forms.
- Lower mining requirements compared to a mined geologic repository may be a secondary advantage.

6.1.2.4 Impacts of Construction and Operation (Preemplacement)

Potential environmental impacts of a rock melt repository would be similar in many respects to those of a mined geologic repository. Both would require surface and subsurface activities that lead to environmental impacts. This impact analysis focuses on unique aspects of the rock melt concept, and refers to discussions on mined geologic emplacement in Section 5.4 as appropriate.

Health Impacts

Health studies related to the rock melt concept for the disposal of HLW can be divided into two phases: the presealing phase, which includes waste transportation and active operation of the waste disposal facility, and the postsealing phase, which includes the melting and resolidification of the HLW/ rock matrix and its long-term effects. In the following discussion, radiological and nonradiological concerns for the first phase are covered separately.

Radiological Impacts. During presealing operations, waste in solution or slurry form would be introduced directly into the repository cavity. Various operations in this charging phase could lead to release of radioactive material into the environment.

Under normal operating conditions, the casing in the emplacement well should prevent contact of radioactive waste with any aquifers that would overlie the disposal cavity. During waste charging, however, it would be possible that some radioactivity could migrate out of the cavity and into the surrounding rock. This possibility would be reduced if the cavity were maintained approximately at atmospheric pressure. Under these conditions, the tendency of water under hydrostatic pressure to flow into the cavity would minimize the importance of this transport mechanism. Nevertheless, it would be possible for radioactive material to reach man through such migration into the surrounding rock and onto the biosphere.

Operational impacts would vary somewhat, depending on which version of the rock melting concept is considered. If liquid HLW were emplaced directly into a cavity from the processing facility, there would be no impacts due to transportation of the waste. If solid waste were slurried into the repository, impacts of waste transportation from the reprocessing plant to the repository would have to be considered. However, such transportation would have no different environmental effects than would the shipping of such wastes to any other type of repository.

Treatment of HLLW prior to emplacement might be required to enhance the compatibility of the liquid with the rock in which the cavity would be located. This additional treatment step would increase the probability of occupational and population exposures to radiation. Handling and treatment of solidified HLW would also increase the probability of radiation exposure; risk analysis would take into account the details of the required handling and treatment procedures.

A summary of potential radiological health impacts was prepared for the rock melting concept (Bechtel 1979a). This study projected the short-term occupational impacts for a single rock melting cavity, which are presented in Table 6.1.8. For a 5,000 MTHM/yr throughput, it is estimated that three rock melting cavities would be required and that the impacts would be linear (Bechtel 1979a). Occupational impacts prior to the waste reaching the repository, nonoccupational impacts, and impacts from abnormal conditions were also postulated in this study. For this analysis, the consequence of impacts under abnormal conditions was found to be comparable to, or slightly less than, those of the other options. This study, however, did not include any probability analysis and consequently total radiological impacts under abnormal conditions have not been quantitatively determined.

Nonradiological Impacts. The underground portion of rock melt repositories would probably be constructed using conventional mining and drilling techniques. Health impacts would be those typical of any analogous construction project, and would be somewhat dependent on the method chosen (whether the cavity were created by mining, underreaming, explosive springing, etc.).

TABLE 6.1.8. Occupational Dose Estimate During Normal Operation
At a Single Rock Melting Cavity

Process Unit	Whole-Body Dose, man-rem/yr
Valve Gallery	120
Offgas Recovery	110
Maintenance	50
Decommissioning	30
Support/Overhead	40
Total	350

Impacts from surface construction would be typical of those associated with the construction of any chemical processing plant. Also, impacts similar to those for the mined geologic repository and discussed in Section 5.4 would be expected for this option.

Natural System Impacts

The effects of rock melting on ground-water migration and transport of radioactivity in the surrounding rock and the possible modeling of these effects are discussed in Section 6.1.2.3. This analysis suggests that heat from the wastes should not affect the thermal regime near the surface.

The principal impacts on natural systems associated with HLW disposal are considered to be those normally encountered in underground drilling and construction activities. Construction impacts could be estimated relative to those from conventional repositories on the basis of the amount of excavation required.

Such topics as disposal of mined spoil, emissions from machinery used in construction, and prevention of water pollution from mud pit overflow could best be analyzed for a specific site. General impacts, however, would be similar to those discussed in Section 5.4.

Because of the lack of formal studies, the effects of the melting cycle on the integrity of the geologic formation would need to be thoroughly studied. Effects such as thermal expansion and contraction, phase change, and hydrologic change during pre- and postemplacement environments would have to be assessed. These effects could be significant, but present data are insufficient to draw meaningful conclusions.

Socioeconomic Effects

Overall, the potential socioeconomic impact of a rock melt repository is rated as minor (Bechtel 1979a). This conclusion is reached, in part, because only a moderate sized work force (between 2,000 and 3,000 people) would be required for successful operation. Land requirements would be less than for any of the other disposal alternatives studied (Bechtel

1979a). In addition, with collocation of three rock melting cavities and three reprocessing facilities at each site, only two facility site locations would be required. The resultant fiscal impact on community facilities would therefore be relatively small.

Although rock melt might have the least socioeconomic impact of any of the alternatives, it is impossible to fully address the nature and extent of impacts at the generic level. This is particularly true when analyzing the socioeconomic impact of construction activity--a detailed estimate of the construction work force has not been completed. Nevertheless, it is reasonable to conclude that socioeconomic impacts would be similar to, and generally slightly less than, those described in Section 5.6 for the mined geologic repository. A cautioning note, however, is that collocation of facilities could lead to a concentration of impacts.

Aesthetic Effects

Facilities associated with a rock melt repository would have an aesthetic impact. The extent of this impact would depend on characteristics at the site and would reflect the fact that optimal engineering design would be necessary for different forms of HLW. Facility design would be a function of the physical and chemical form of the HLW.

The extent of surface construction would depend on the rock melting concept version for which the repository was being designed; where HLW solutions were being directly emplaced, the entire reprocessing plant would be located close to the repository. Where waste slurries were emplaced, only a relatively simple surface installation would be required to condense steam, add makeup water, provide for slurry mixing, etc. Aesthetic impacts would reflect final facility design, with larger facilities generally having greater impacts. Overall, aesthetic impacts would be similar to those described for a mined geologic repository, as presented in Section 5.6, with minor exceptions.

Facilities that would be different from those in the mined geologic repository include the type of cooling towers and tall drill rigs used in excavating the rock cavities. In addition, although a 100-m-high stack would be required for a processing facility, its location on the same site as the repository would reduce overall aesthetic impacts. Other aesthetic impacts, such as noise and odor, have not been identified as a problem with rock melt.

Resource Consumption

Energy would be required to construct and operate a rock melt disposal system. Initially, energy would be consumed in transportation and construction activities. In the operational phase, waste preparation, transportation, and emplacement activities would consume energy. Quantitative estimates of energy consumption for the construction and 40 year operation of a 5,000 MTHM/yr system have been prepared (Bechtel 1979a). These estimates are presented in Table 6.1.9.

Consumption of other critical materials has not been identified as an important factor in evaluating the merits of the rock melt concept. Drilling activities, as well as construction of the facilities, would require steel, cement, and other construction materials typically associated with a major facility. Estimates of these requirements are presented

TABLE 6.1.9. Estimated Energy Consumption (Bechtel 1979a)

Propane, m ³	1.0 x 10 ⁶
Diesel, m ³	1.5 x 10 ⁶
Gasoline, m ³	1.5 x 10 ⁵
Electricity, kWh	5.7 x 10 ¹⁰

in Table 6.1.10 (Bechtel 1979a). No scarce or otherwise critical material has been identified as being important for this option.

As noted, the reference concept calls for each rock melting repository site to support three 6,000 m³ cavities about 2,000 m below the surface (Bechtel 1979a). Each site would be able to accommodate waste from 5,000 MTHM/yr for 25 years. Construction of these facilities would disturb 1,100 hectares (2,720 acres) of land and would require a restricted land area of 4,000 hectares (9,880 acres) (Bechtel 1979a). Most of the land disturbed would be required for processing, encapsulation, and other surface facilities.

International and Domestic Legal and Institutional Considerations

The rock melting concept would have relatively few international implications because waste transportation activities would occur in the U.S. and emplacement would be achieved well out of range of the biosphere. There are, however, important domestic legal and institutional considerations that would need to be resolved. For example, as noted in Section 6.1.2.2, retrieval of wastes, even before emplacement activities were complete, would be very difficult. The hot nature of the wastes and the type of waste packaging that would be employed would influence the ease with which the waste material could be withdrawn. Retrieval after the cavity was sealed and the waste was in a molten form would be impossible. Legal and regulatory implications of these restrictions on retrieval would have to be resolved.

Selection of the rock melting concept would also affect certain decisions regarding interim storage. If waste from the uranium-only recycle, or the uranium and plutonium recycle were stored, it would be necessary to specify the form of waste storage that would have the least environmental and economic impact. Although it is possible that the waste

TABLE 6.1.10. Estimated Material Consumption (Metric Tons)

Carbon steel	300,000
Stainless steel	24,000
Components	
Chromium	4,800
Nickel	2,200
Tungsten	--
Copper	1,900
Lead	2,900
Zinc	600
Aluminum	900

would be stored as a liquid, it is more probable that it would be solidified (calcined or vitrified) if an extended storage period were envisaged.

6.1.2.5 Potential Impacts Over the Long Term (Postemplacement)

Although repository-related human activity would be minimal once emplacement and repository decommission activities were complete, impacts could occur because of the possible mobility of the molten waste material in the geologic environment. Potential events and impacts are described below.

Potential Events

For risk analysis purposes, the postemplacement phase of the concept is treated in a manner similar to other geologic disposal alternatives (see Section 5.6). As noted earlier, after the waste-rock matrix cooled to the point where liquid water could contact the waste, it is assumed that the waste would dissolve, and transport through the surrounding rock would be initiated. Clearly, the degree of risk calculated on this basis would be strongly site specific, and would depend on factors such as the depth of the repository, presence and location of aquifers, water quality, and sorptive properties of the rock.

Possible pretreatment of the wastes to minimize potential adverse postemplacement effects would depend on the waste form as well as the geologic media characteristics.

Potential Impacts

Basically, the environmental considerations involved in evaluating the long-term impact of rock melting are how much of the radioactivity in the repository would reach the biosphere, when it would get there, and what its effects would be.

The heat barrier effect is discussed in Section 6.1.2.3. Following total resolidification (1000 years), when the heat barrier no longer existed, most fission products would have decayed to innocuous levels. The toxicity of the residual radionuclides in the resolidified waste-rock matrix at that time should be significantly less than that of a typical uranium ore body from which the nuclear fuel was originally extracted.

Mixing of the HLW with the molten rock, as well as the physical and chemical properties of the cooled and resolidified waste-rock matrix, would determine the rate at which radioactive species could be leached and transported by ground water. It might be possible to design some mitigating measures to significantly retard leaching rates of all or some of the radioactive species present.

It is possible that the heat barrier effect would retard the start of effective leaching of radioactivity until radioactive decay had essentially eliminated the fission products as significant health hazards; thus, it might be necessary to consider only the TRU products.

Transportation of radioactivity by ground water would have to be evaluated on a site-specific basis, although different scenarios could be postulated to obtain order-of-magnitude estimates of the time required for radiation to appear in the biosphere and of the concentrations of radioactive species that would be present in the water. In modeling the

radioactivity transport, movement of water would be considered as taking place both through permeable rock and by means of joints and cracks in low-permeability rock (Heckman 1978). The impacts of a ground-water breach of a rock melt repository are expected to be similar to those that would result if a mined geologic repository were breached by ground water (Bechtel 1979a).

6.1.2.6 Cost Analysis

Cost estimates for the rock melt concept do not have the benefit of a reference conceptual design, nor of previous cost estimates for similar types of facilities. Therefore, these cost estimates are only approximate. They are based on the reference concept disposal of HLW from 5,000 MTHM/yr, for 25 years, requiring three cavities.

All cost estimates are in 1978 dollars based on January 1979 dollar estimates (Bechtel 1979a) less 10 percent.

Capital Costs

The capital cost of a rock melt repository with an operating lifetime of 25 years is estimated at \$560 million.

Operating Costs

An allowance of 2 percent of the capital cost is assumed for the annual operating cost, which comes to \$11 million a year.

Decommissioning Costs

The total decommissioning cost for the three-cavity rock melting concept is estimated at \$21 million. In this estimate, final shaft sealing is treated as a decommissioning cost with an allowance of \$2 million per cavity.

6.1.2.7 Safeguard Requirements

Because of the restrictions concerning the transportation of radioactive liquids, the fuel reprocessing plant would have to be collocated with the rock melt repository. Therefore, accessibility to sensitive materials would be extremely limited with liquid emplacement. If the waste were to be placed in a solid form (e.g., pellets), which could be emplaced in the subsurface cavity as a slurry, the fuel reprocessing plant could be located off site but transportation related safeguards would then be required. The subsurface cavity would increase the difficulty of diversion and the liquid or slurry waste form would complicate the transportation and handling problems for potential diversion. However unlikely, retrieval by drilling and pumping is possible. This would eventually need to be considered for rock melt repository safeguards. Material accountability would be enhanced by ease of sampling and measurement, but gross accountability (i.e., gallons vs. canisters) would be slightly more difficult than for the mined geologic repository concept. For additional discussion of predisposal operation safeguards see Section 4.10.

6.1.3 Island Disposal

6.1.3.1 Concept Summary

Island-based disposal would involve the emplacement of wastes within deep, stable, geological formations, much as in the conventional mined geologic disposal concept discussed in Chapter 5 with an over-water transportation route added. The island would provide port facilities, access terminals, and a remote repository location with possibly advantageous hydrogeological conditions. An island disposal facility could also provide an international repository if the necessary agreements could be obtained.

The island disposal concept has been referred to as an "alternate geologic approach" (Deutch 1978) in which the geology (i.e., rock, sediments) provides the primary barrier between the nuclear wastes and the biosphere and the ocean may provide an additional barrier, depending on the repository location and the hydrological system existing on the island.

The status of the concept is uncertain. The U. S. Department of Energy Task Force Draft Report (Deutch 1978) stated that "The Department of Energy has no program to actively investigate the concept. Suggestions for assessment of the concept have been made from time to time by groups considering international aspects of radioactive waste repositories. However, a consensus for the need of such repositories has not developed."

On the other hand, the sixth report of the U. K. Royal Commission on Environmental Pollution (Flowers 1976) referred to island locations when considering hard rock sites for a geologic facility. In this report, it was stated that "A deep disposal facility on a small uninhabited island would be particularly advantageous if one were chosen which was separated hydrogeologically from the mainland. Any leakage of radioactivity into the island's ground water would be easily detected and in that event the dilution of seawater would provide a further line of defense."

No detailed studies of the island concept are currently available; therefore, its basic elements are based on simplified modification and adaptations of conventional mined geologic disposal as discussed in Chapter 5. Since the geology of most islands is crystalline rock, it is the assumed disposal formation. Elements of other schemes (e.g., subseabed disposal, Section 6.1.4) have been incorporated and/or referenced where appropriate. If more detailed assessments are required in the future, conceptual design studies would have to be performed to provide a reliable basis for analysis.

6.1.3.2 System and Facility Description

System Options

The reference concept for the initial island disposal of nuclear waste has been developed from a number of options available at each step from the reactor to disposal in the island geology.

Various options to be considered for island disposal are summarized in Figure 6.1.7, with options for the reference concept designated. Details on the bases for selecting reference concept options are covered in various documents listed in Appendix M.

Because system options for island waste disposal beginning with the reactor and including steps up to the transportation requirements are similar to those for mined geologic repositories, the options selected for the reference design are similar for the two concepts. From that point on, the selected options are based on current program documentation.

Waste-Type Compatibility

An island repository could handle all wastes from the uranium and plutonium recycle case, and from the once-through cycle.

Waste-System Description

The reference island repository design is based on the concept discussed in Section 6.1.3.1 and the waste disposal cycle options identified above. The fuel cycle and process flow for the reference concept are shown in Figure 6.1.8. The reference system assumes the transport of all spent fuel, HLW and transuranic wastes to the island sites.

The waste forms and emplacement concept of canistered waste for island disposal would be the same as those for conventional mined geologic disposal discussed in Chapter 5.

Predisposal Treatment and Packaging. The predisposal treatment of waste for the island disposal concept would be identical in most respects to the predisposal treatment of waste for mined geologic repositories. Chapter 4 discusses the predisposal systems for both spent fuel and HLW common to all of the disposal concept alternatives.

Geologic Environments. The geohydrologic regime of an island, as diagrammed in Figure 6.1.9, comprises a self-contained freshwater flow system (called the freshwater lens because of its general shape), floating on a sea-fed, saline ground-water base. There are two possible locations for the repository--in the lens of freshwater circulation and in the deep, near-static saline ground water - shown as A and B in the figure.

Geographically, three classes of island have been identified:

- Continental Islands - located on the continental shelves and including igneous, metamorphic, and sedimentary rock types
- Oceanic Islands - located in ocean basins and primarily of basaltic rock of volcanic origin
- Island Arcs - located at margins of oceanic "plates", primarily of tectonic origin, and frequently active with andesitic lavas.

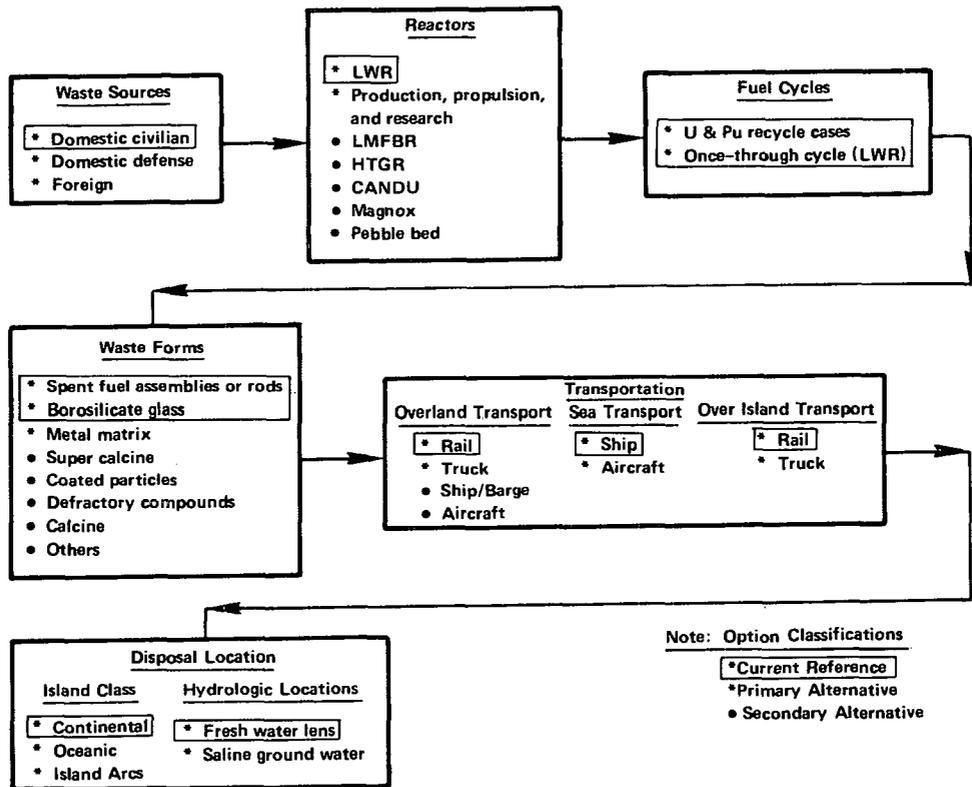
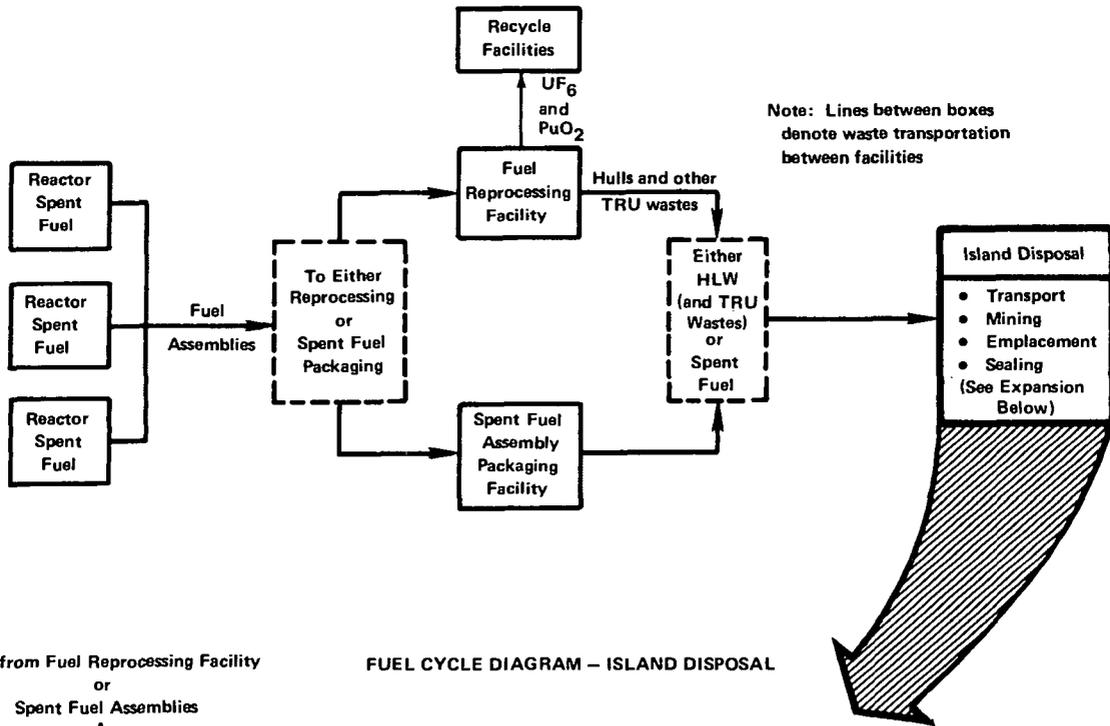


FIGURE 6.1.7. Major Options for Island Disposal of Nuclear Waste



HLW from Fuel Reprocessing Facility
or
Spent Fuel Assemblies

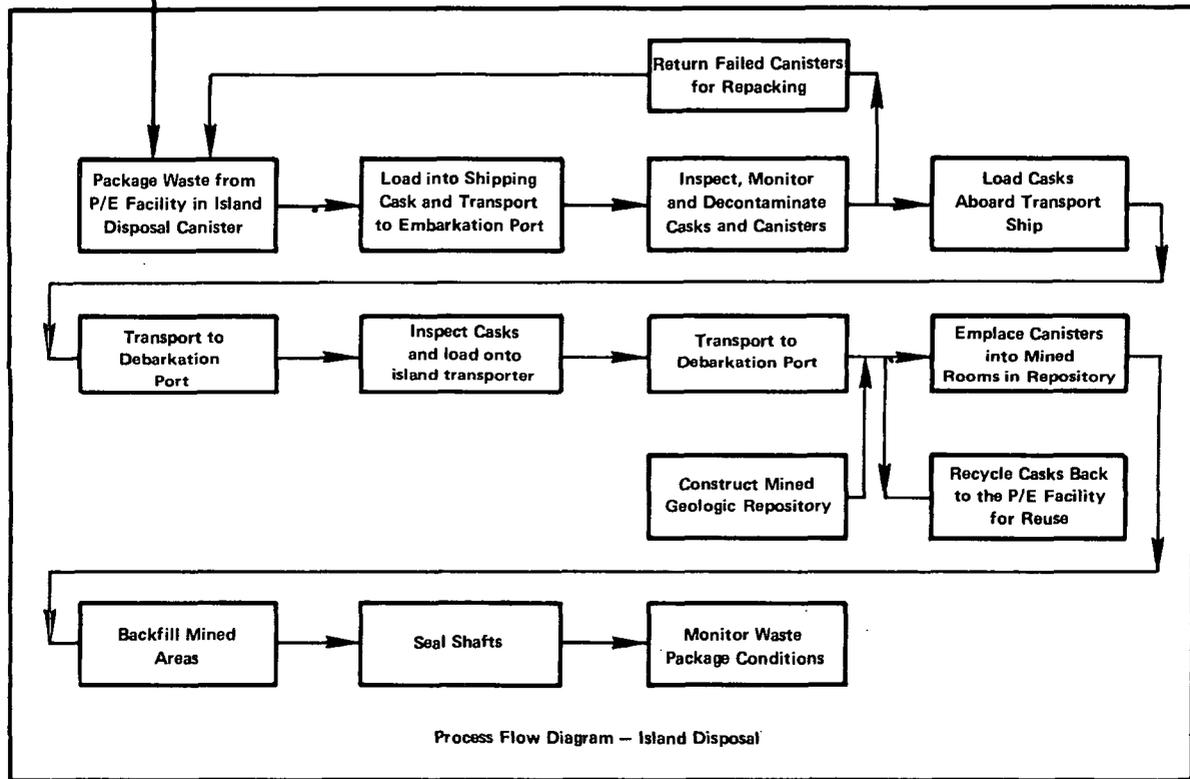


FIGURE 6.1.8. Waste Management System--Island Disposal

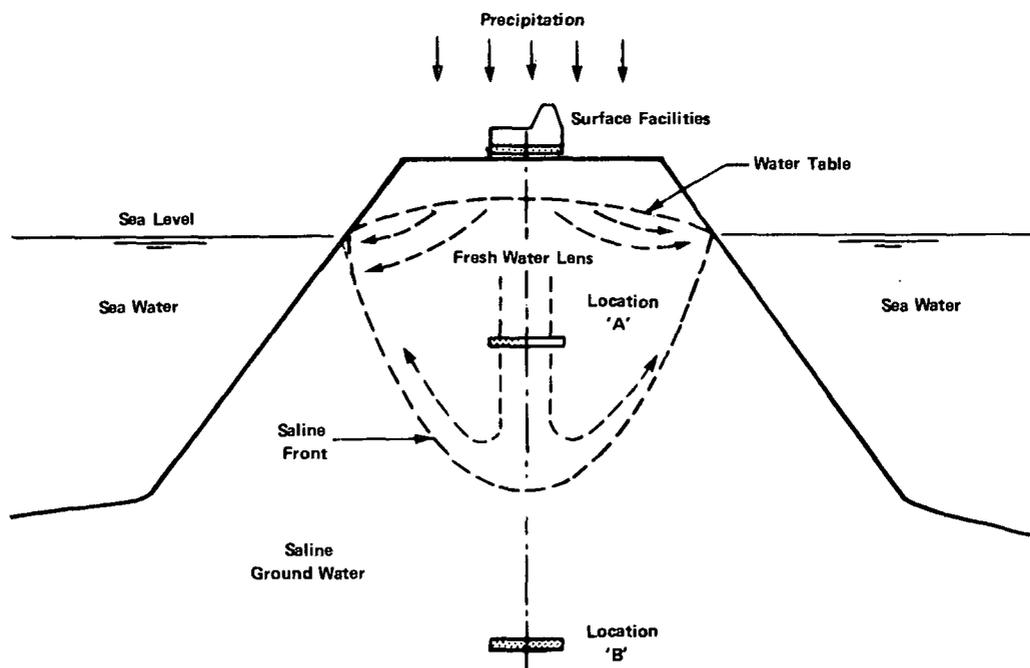


FIGURE 6.1.9. Hydrological Classification of Repository Locations

All three classes exhibit the classical island geohydrology described above, as modified by local geology and geographic setting. There are further discussions of the geology and hydrology of typical islands in DOE (1979), Todd (1959), Bott (1971), and Bayley and Muehlberger (1968).

Transportation Features. The island concept would incorporate the same basic procedure for transportation and handling as mined geological disposal. Of course, additional transportation from the mainland port to the island and additional receiving and handling facilities would be required. Transportation from the fuel reprocessing plant to the disposal site would be accomplished in three stages. The first stage would consist of truck or rail transport to a mainland port. Waste would be carried in transport casks that would cool the wastes and provide radiation shielding. (See Chapter 4 for a discussion of this procedure.) The second transport stage would be by ship to the island port. The subseabed disposal option (Section 6.1.4) details the operational features of this transportation phase. The casks would be cooled by either a closed-circulation water system, filtered forced-air system, or heat exchangers cooled by seawater. The coolant would be continuously monitored for radiation and temperature changes. Ship construction would provide for additional cooling. The ships could also include a shielded cell facility for examination of the casks.

The receiving port at the island would have the same features as the embarkation port described in Section 6.1.4. It could have a facility for temporary waste storage and transfer of the waste to specially designed transportation casks for final transport to the repository, the third phase. Conceptual design studies for island disposal are unavailable, but the required additional transportation facilities might be based on those discussed for the port and sea transport parts of the subseabed disposal option in Section 6.1.4.

Repository Facility. The layout of the reference repository for island disposal is a preliminary adaptation of the conventional geologic disposal concept discussed in Chapter 5. It is assumed that the island bedrock is crystalline and that the waste is emplaced approximately 500 m underground.

The conceptual design for an island crystalline rock repository is not supported by a data base comparable to that for salt repositories. The crystalline rock conceptual design discussed in Chapter 5 is assumed to be applicable to the underground aspects of island disposal except salt stockpile handling equipment would not be needed. The surface facilities for island disposal are assumed to be the same as for conventional mined geologic disposal.

Assuming that the repository capacity for spent fuel disposal is the same as for the conventional mined geologic disposal and that sufficient intermediate storage and transportation capacity can be provided, the once-through cycle would require four to eight island repositories, depending on the media. More repositories would be needed if island area were insufficient to support a repository of the size discussed in Chapter 5. Uranium-plutonium recycle wastes would require six to ten island repositories, depending on the island media (DOE 1979). The scheduled availability of the repositories for wastes from both fuel cycles would be expected to be a few years behind that of the conventional mined geologic disposal program.

Retrievability/Recoverability. Retrievability of emplaced waste or spent fuel from the rooms would be essentially the same as for the conventional mined geologic repository in crystalline rock. If retrieval were required because of deterioration or failure of the waste containers, special transportation containers and storage facilities would be needed. This need could be met by using a special cask design suitable for either rail, truck, or sea transport. Recoverability would also be similar to that with mined geologic disposal and would involve techniques similar to those used for the original emplacement process. Retrievability from island repositories could be complicated by the hydrogeologic characteristics of the sites.

Sealing, Decommissioning, and Monitoring. The sealing concepts might be the same as those for conventional mined geologic disposal in crystalline rock. The principal difference would be in the supply of labor and materials, which would involve sea transport to the island.

Final decommissioning of the island facilities could involve underground disposal of all contaminated equipment, the removal or disposal of all surface facilities, and suitable restoration and landscaping of the island.

Monitoring systems would be used during emplacement operations to detect air, surface water, and ground-water contamination. After the repository was sealed, a long-term monitoring system would be implemented. This system would be similar to those for the conventional geologic disposal concept, with modifications to suit the island option.

6.1.3.3 Status of Technical Development and R&D Needs

Present State of Development

In general, conventional mining techniques would be applicable to island repository construction. Transportation, storage, and handling requirements would be similar to those for the conventional mined geologic disposal concept, with the addition of the sea transportation link. Construction methods for ports would employ standard engineering practice.

Because the island disposal concept is so similar to the mined geologic repository option, the state of development is about the same. The ship loading and unloading requirements are similar to those described in the subseabed alternative, so again, the state of development is about the same.

Technical Issues

Technical issues that differ from those for mined geologic repositories lie in the areas of unique island hydrology and the resultant impacts of fresh or saline water on the package materials and the waste formulation.

For example: Is the waste form proposed for conventional mined geologic disposal appropriate for island disposal? Are the canisters that encapsulate HLW or the canisters of spent fuel compatible with the island repository environment? Should emplacement be in the freshwater zone or the saline ground-water zone?

Because a major incentive for considering island sites is a particular hydrological regime that frequently exists beneath them, efforts would be needed to:

- Verify the existence of a freshwater lens at various sites and determine its size.
- Determine the flow patterns and velocities of saline ground water at depths beneath the freshwater lens.
- Verify the stability of the freshwater lens in terms of the equilibrium between deep groundwater flows, salinity diffusion, precipitation and surface hydrology, the effects of sea level slopes, and other relevant processes in the natural state.
- Examine the perturbation to the lens caused by construction of the repository shafts and underground facilities, using simulation models and field evidence, if available. The shafts and facilities will tend to provide a sump that will drain either the freshwater or the saline ground water, depending on the location and depth of the repository.
- Examine the effects of heat generation on lens stability using simulation models. Heat may cause thermal convection cells that could flow counter to the freshwater circulation and modify the discharge pattern into the seawater.

R&D Requirements

To resolve these technical issues, specific R&D programs would be directed toward:

- Development of a system data base
- Study of hydrogeological aspects of island sites

- Development of criteria for and categorization of siting opportunities
- Risk assessment.

Implementation Time and R&D Costs

The time to complete the R&D, and the associated costs would be very similar to time and costs for a mined geologic repository. Increased R&D cost for the island concept would be expected to be a very small increment when compared to total costs for development of the mined geologic repository.

Summary

Major uncertainties, shortcomings, and advantages of the concept are summarized below:

- The transportation requirements to a remote location add to the overall risk of the concept.
- The state of knowledge relating to the hydrologic regime, upon which the concept relies, is not currently sufficient for siting or performance analysis.
- Considerable effort might be required to develop specialized waste forms and packages, if current reference concepts are not suitable.
- The approach does appear to be technically conservative if the hydrology is as predicted and to be capable of implementation in a step-wise manner.
- The concept employs the multi-barrier approach and has the additional attractive benefit of being remote.

6.1.3.4 Impacts of Construction and Operation (Preemplacement)

Impacts of construction and operation of predisposal systems in the island concept would be similar to those discussed in Section 5.6 for the mined geologic repository. Additional impacts from the sea transportation link and the port facilities would also be involved and are discussed in Section 6.1.4.4 for the subseabed disposal option. Impacts of mainland disposal are not discussed here.

Ideally, any island chosen for disposal would be totally uninhabited prior to construction of the repository (Selvaduray et al. 1979). In this case, the only non-occupational people impacted by construction and operation of the island repository would be families of those working at the facility.

Health Impacts

Radiological Impacts. Increased radiation exposure of occupational personnel under both normal and abnormal conditions would result from unloading of the waste at the receiving port, temporary storage of the waste, and transfer of the waste to the repository. Quantitative estimates of these exposures are not available at this time. However, unloading of the waste would probably result in exposures similar to those encountered during loading at the embarkation port, as discussed in Section 6.1.4.4 for the subseabed option. In addition, it is significant that the island repository would accept TRU wastes. This means that transportation impacts would be slightly greater than those for the subseabed option.

Moreover, although transportation-related impacts might be higher for island disposal, mainland benefits would be significant because of the elimination of the need to dispose of TRU wastes on the mainland.

The operation of the island repository itself is expected to be essentially the same as that for a mined geologic repository. Therefore, the exposure of occupational personnel to radiation should also be essentially the same. This exposure, during both normal and abnormal conditions, is discussed in Section 5.6.

In the event that there were any nonoccupational people on the island, the maximum dose received by any one of those individuals is expected to be similar to that received as a result of the operation of a mined geologic repository. However, because only a limited number of nonoccupational people should be present, total nonoccupational radiological health effects for an island repository are expected to be considerably less than those for a mined geologic repository.

Nonradiological Impacts. As indicated, impacts for island disposal should be similar to those of the subseabed and mined geologic disposal options. However, for an island repository in a relatively uninhabited area of the world, impacts would be significantly different from those of the mined geologic repository. In that case, potential non-occupational impacts would result primarily from transportation activities. Most transportation-related impacts are expected to be similar to those from the subseabed disposal option and are described in Section 6.1.4.4. That option, however, would not involve unloading waste material and increased transportation that could cause additional impacts from island disposal.

Natural System Impacts

Investigation of candidate island disposal sites would involve drilling and geophysical surveys, both on the island and in the adjoining offshore areas. During these activities, natural and wildlife habitats could be disturbed. Access and exploration operations could pollute both freshwater and seawater sources. Ecological effects could also arise from the use of explosives for seismic surveying. These impacts could be minimized by identification of sensitive areas and adequate planning.

Other ecological impacts, such as those described for the mined geologic repository in Section 4.8, would occur on the island selected for final disposal. However, because of the delicate balance of an island ecosystem, these impacts might require special consideration. In addition, the construction and operation of the required transportation and repository facilities would potentially impact the marine environment. These types of impacts have not been extensively evaluated.

Another important consideration is that small island ecosystems provide no refuge for the biota and ecosystems are much more easily affected by large-scale human activity. Furthermore, after the operational phase had ended, recolonization from outside sources would be far more difficult, and would take longer, than for a continental region. Finally, the types of

species that recolonize an island could be expected to establish considerably different trophic structures than were present prior to construction.

Emplacement operations in the repository would be similar to those for the conventional mined geologic disposal concept. However, if an accident were to occur within the island repository, water might be present because of drainage into the excavation. Thus, these operations, and other activities associated with the island repository, could affect the freshwater regimes on the island. In addition, water pumped from the underground excavation would be brackish if the repository were located below the freshwater lens in the saline zone. Therefore, care would be required to prevent contamination of surface freshwater streams and lakes. Disturbance of the natural ground-water regime could result in some freshwater wells becoming saline. Such activity could significantly affect the island's ecosystem, of which freshwater is a critical element.

Socioeconomic Impacts

Construction of an island repository would require assembling and transporting a large work force to a remote island. These activities would affect the socioeconomic structure of coastal communities through which the project personnel and equipment were transported. Detailed assessment of these impacts has been limited, but information presented on the subseabed and ice sheet options provides a useful perspective (Sections 6.1.4.5 and 6.1.5.5).

On the island, socioeconomic impacts would be a different type of concern associated with the entirely new communities that would normally be established. Selecting unoccupied islands for a final repository would greatly reduce socioeconomic impacts.

Aesthetic Impacts

Aesthetic impacts of the island disposal option would be limited because few people would live in the vicinity of the repository. During construction and operation, authorized site personnel would be the only individuals to perceive aesthetic impacts.

Aesthetic impacts would also be associated with transportation activities. Although these are generally not viewed as significant, additional discussion on this matter appears in Sections 6.1.4.5 and 6.1.5.5 on the subseabed and ice sheet disposal options, respectively.

Resource Consumption

Construction and operation of the island repository facilities would require energy, as would transporting the waste material to the disposal site, over mainland, ocean, and island routes. There are no studies available to quantify these energy needs.

Although the size of the facility and the land area required would be similar to that for the conventional mined geologic concept, it should be recognized that island repositories would likely require that an entire island be devoted to a waste repository. This commitment of land might not be important, however, considering that extensive study would be completed before an individual island was proposed as a disposal site.

International and Domestic Legal and Institutional Considerations

The island disposal option, like the subseabed and ice sheet options, would require transporting waste material over the ocean, and the general international implications of such transportation are important. Emphasis in this discussion is placed on aspects unique to island disposal.

Two, possibly complementary, international considerations would have to be studied for island disposal. On the other hand, an initial motivation for island disposal is that it could provide an international repository for use by many countries. On the other hand, the siting of a repository on an island over which the U.S. does not have sovereignty would require the approval of the nation that does.

International concerns could arise from countries in the vicinity of a proposed island repository. For example, if a remote island in the South Pacific were selected for an island repository, nations bordering the South Pacific might feel they were exposed to risks while receiving little or no benefit. Regardless of whether specific treaties were required, nations adjacent to any island disposal site would be likely to voice concern and seek international assurance of the safe operation of these facilities.

6.1.3.5 Potential Impacts Over Long Term (Postemplacement)

Potential Events

As in land disposal of radioactive waste, island disposal would require careful assessment of the processes by which the radionuclides could migrate from the containers through the various barriers to man's environment. Actual island emplacement of any quantity of such waste could occur only after the completion of a program to demonstrate, by analysis and experiment, the retention capabilities of each of the natural and man-made barriers to migration.

Waste Encapsulation. The waste form and canisters used for island disposal might be similar to those used in a mined geologic repository on the mainland. Studies of the specific effects of ground-water chemistry in either the freshwater lens or deep saline zones would provide data for establishing leach rates in the crystalline rock site.

Ground-Water Transport, Freshwater Lens Location. Waste emplaced in the freshwater lens might be exposed to the very slow ground-water circulation within the lens. The velocities would depend on rock permeabilities, porosities, precipitation, and surface hydrology. A simplified conceptual view of the potential pathways and barriers is shown in Figure 6.1.10.

Waste in the freshwater lens circulating system might be expected to discharge at the shoreline. Natural ground-water flow patterns might be affected by thermal convection and repository construction. Concentrations at the exit zone have not been estimated.

Radionuclides might be sorbed by the host rock, which would substantially retard the waste transport within the lens. Sediments that might exist at the shoreline in the discharge zone could have useful sorption properties and retard radionuclides prior to discharge and dilution in the seawater.

Ground-water Transport, Saline Zone Location. It has been suggested that offshore islands may have essentially static saline ground water at depth, due to the absence of hydraulic gradients at sea level. However, the residual or continuing effects of oceanographic, geothermal, climatological, or other changes may create flow. These effects would need to be examined prior to siting a repository in such a location (see Figure 6.1.11).

Flow transport in the saline zone may be accompanied by dispersion and diffusion, which would result in reduced concentrations at a distance from the repository. The amount of sorption of radionuclides in the host rock or on seabed sediments would depend on the particular radionuclide, ground-water, and rock or sediment chemistry.

Seawater Contamination. It appears that the principal discharge of wastes from an island repository would be into the seawater, possibly through sediments. Discharge might occur in a relatively concentrated near-surface zone if the waste were located in the freshwater lens. This could cause contamination of littoral and near-surface aquatic systems.

Discharge from wastes located in the saline ground-water zone would likely be dispersed through the seabed if the thermal-convection effects were insufficient to distort the flow patterns significantly.

Volcanism. Some islands, particularly those in island arcs and to a lesser extent oceanic islands, are frequently highly active seismically and volcanically. Such activity could discharge the waste in either lava flows or into the atmosphere. Geologic data for the most recent volcanic event would be relied upon to establish inactivity before an island was selected as a disposal site.

Potential Impacts

In determining the potential impacts of island disposal over the long term, the following factors would be considered:

- Corrosion, leaching, and transportation of radionuclides to the biosphere by the ground water
- The influence of thermal effects on flow
- Thermal/mechanical effects on permeability and porosity
- Retardation of radionuclides on rock fractures and seabed sediments
- Sediment and current movements
- Pathways to man via marine organisms, typical marine activities, and island considerations.

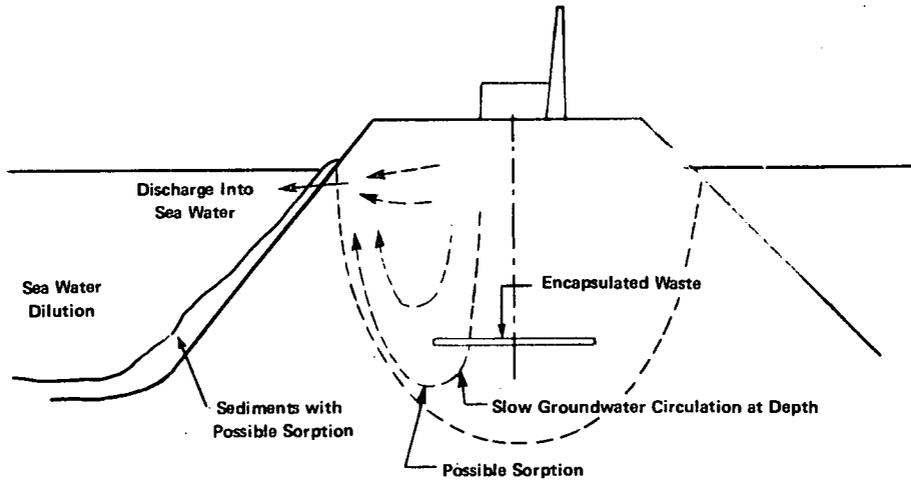


FIGURE 6.1.10. Isolation Barriers for Freshwater Lens Location

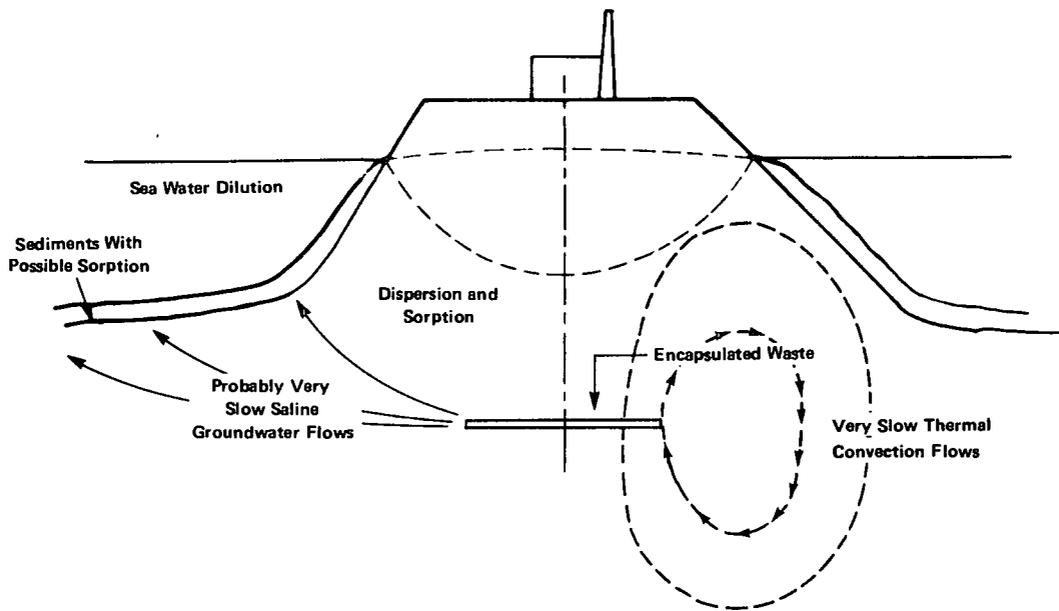


FIGURE 6.1.11. Isolation Barriers for Saline Zone Locations

Quantitative estimates of these impacts for the island disposal concept are unavailable at this time. However, it is expected that they would be similar to, but probably less significant than, those from a mined geologic repository. The reasons for the probable lessened impact are that (1) seabed sediments might provide significant sorption of certain radionuclides, (2) the sea would provide substantial dilution of discharges from the ground water, and (3) the island population, which would bear the greatest impacts, would be expected to be small in the long term because of the remoteness, size, and limited potential for inhabitation of any island that would be selected.

6.1.3.6 Cost Analysis

Detailed costs for island repository construction, operation, and decommissioning have not been estimated. It is estimated, however, that the cost of an island repository would be at least double that for a continental mined geologic repository because of sea transportation, the associated loading and unloading facilities, and the high salaries necessary for remote locations.

6.1.3.7 Safeguard Requirements

With the exception of ocean transportation, safeguard requirements for this concept would be expected to be similar to those for the mined geologic repository concept. However, the risk of diversion for the island disposal concept is primarily a short-term concern because of the remoteness of the disposal site and the major operational and equipment requirements for retrieval. Physical protection of the sensitive facilities and transportation operations would be the most effective way to deny access to the waste for the short term. For additional discussion of predisposal operations safeguards see Section 4.10.

6.1.4 Subseabed

6.1.4.1 Concept Summary

In subseabed disposal, wastes would be emplaced in sedimentary deposits of the ocean bottom that have been stable for millions of years. These deposits have a high sorptive capacity for the waste species (except for iodine and technetium) that might leach from the waste packages. Transport from ocean depths for any waste species escaping the sediments to the biologically active near-surface waters is expected to be a slow process that would result in dilution and dispersion. In addition, the great depth of the water column would constitute a barrier to human intrusion.

A program has been under way since 1973 to assess the technical and environmental feasibility of this concept for disposing of high-level nuclear wastes (Bishop 1974-75, Talbert 1975-78). The total seabed represents about 70 percent of the surface of the planet (of which less than 0.0001 percent would be used) and contains a wide variety of geologic formations. Theoretically, all wastes from the once-through cycle and uranium-plutonium recycle options could be emplaced in subseabed formations. But, because of volume considerations, other methods of disposal may be more practicable for contact handled and remotely handled TRU wastes.

The reference subseabed geologic disposal system for study purposes is the emplacement of appropriately treated waste or spent reactor fuel in a specially designed container into the red clay sediments away from the edges of a North Pacific tectonic plate, under the hub of a surface circular water mass called a gyre (mid-plate/gyre:MPG). (However, selection of the North Pacific as a study area in no way implies its selection as a candidate subseabed disposal site.) The reference method uses a penetrometer^(a) for emplacing wastes in the sediments in a controlled manner that allows subsequent monitoring. A specially designed surface ship would transport waste from a port facility to the disposal site and emplace the waste containers in the sediment. A monitoring ship, which would completely survey the disposal site before operations began, could determine the locations of individual disposal containers and monitor their behavior for appropriate lengths of time. The ship would also maintain an ongoing survey of the surrounding environment.

(a) A penetrometer is a needle-shaped projectile that, when dropped from a height, penetrates a target material. It can carry a payload of nuclear waste and instruments designed to measure and transmit its final position and orientation relative to the sediment surface. Penetration depth is controlled by the shape and weight of the penetrometer, its momentum at contact with the sediment, and the mechanical properties of the sediment.

6.1.4.2 System and Facility Description

System Options

The reference concept for the initial subseabed disposal of nuclear waste has been developed from a number of options available at each step from the reactor to disposal in the subseabed repository.

Various options to be considered for the subseabed concept are summarized in Figure 6.1.12. The bases for selection of options for the reference concept are detailed in sources cited in Appendix M.

Waste-Type Compatibility

It is assumed for the reference case that subseabed disposal is limited to disposing of spent fuel, HLW and cladding hulls. Other wastes are assumed to be disposed of in a mined geologic repository. However, it should be noted that these wastes may also be appropriate for subseabed disposal if there are sufficient economic incentives.

Waste-System Description

The reference concept design was selected as a feasible approach based on available information and data and is not supported by a detailed system engineering or cost analysis. The waste-management system, including the fuel cycle and process flow, for the reference concept is shown in Figure 6.1.13.

Subseabed disposal has as its foundation a set of multiple barriers, both natural and man-made, that would be employed to ensure the safe isolation of nuclear waste. These barriers are (Bechtel 1979a):

- The waste form
- The waste canister
- The emplacement medium (i.e., sediment)
- The benthic boundary layer
- The water column.

The water column is a barrier primarily to intrusion by man, although it would provide dilution and dispersion for radioactive species.

The waste form (leach-resistant solid) and the metallic waste canister or overpack would be man-made barriers. It is assumed that they could be engineered as a multibarrier system to contain the waste for a period during which the heat-generation rate due to fission product decay would decrease to low levels.

The emplacement medium (clay sediment) shows evidence that it could provide long-term containment of the nuclides through its sorptive qualities, ion-exchange characteristics, and very low permeability.

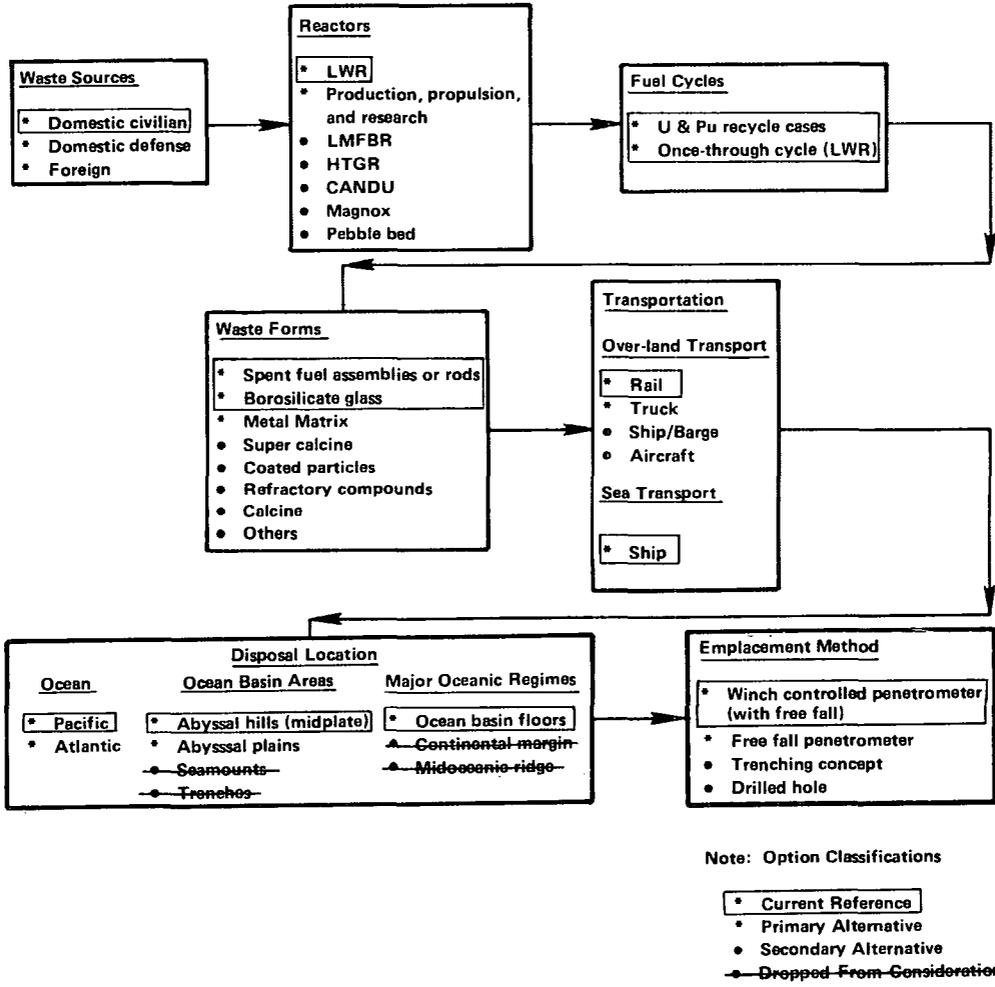


FIGURE 6.1.12. Major Options for the Subseabed Disposal of Nuclear Waste

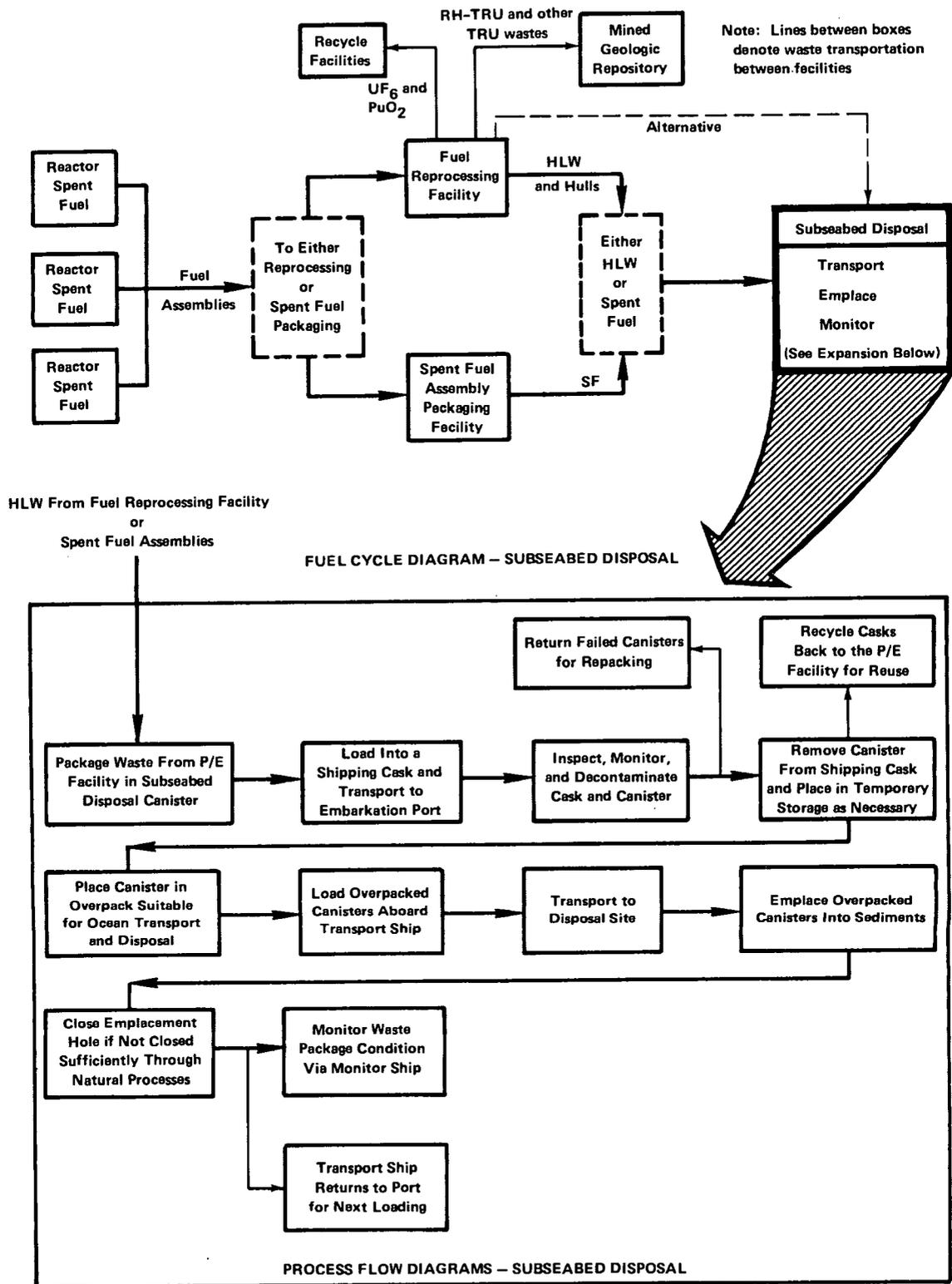


FIGURE 6.1.13. Waste Management System--Subseabed Disposal

The ocean's benthic boundary layer extends from less than 1 m below the sediment-water interface to 100 m above that interface. This layer results from the turbidity induced by natural flow processes and by the biological activity at, or just below, the sediment-water interface. Particulate matter, which would act to sorb radionuclides escaping the sediments, is temporarily suspended in this layer and then returns to the sediment surface.

The water column extends from the benthic boundary layer to the surface of the water. It would provide dilutional mitigation to the release of radionuclides. It would also be a barrier to man's intrusion.

Predisposal Treatment. The predisposal treatment of waste for the subseabed concept would be identical in many respects to the predisposal treatment of waste for the mined geologic repository concept. Chapter 4 of this document discusses the predisposal systems for both spent fuel and HLW common to all of the disposal concept alternatives.

Ocean Environment. Analysis of ocean regimes has shown that the most appropriate areas for subseabed waste containment would be clay-covered abyssal hill regions away from the edges of subocean tectonic plates underlying large ocean-surface currents known as gyres. These vast abyssal hill regions are remote from human activities, have few resources known to man, are relatively biologically unproductive, have weak and variable bottom currents, and are covered with red clay layers hundreds of meters deep.

These clay sediments are soft and pliable near the sediment-water interface and become increasingly rigid with depth. Tests have shown that they have high sorption coefficients (radionuclide retention) and low natural pore-water movement. Surface acoustic profiling indicates that such sediments are uniformly distributed over large areas (tens of thousands of square kilometers) of the ocean floor. As shown by core analysis, they have been continuously deposited and stable for millions of years, giving confidence that they would remain stable long enough for radionuclides to decay to innocuous levels (DOE 1979).

Transportation Features. The overland transportation features of the subseabed disposal concept would be essentially identical to those of the mined geologic disposal concept. In addition, subseabed disposal would require transportation of the waste from the mainland to the subseabed repository. The principal transportation requirements would be for seaport facilities and seagoing vessels.

a. Seaport Facilities. The subseabed reference concept assumes that seaport facilities would be used only for waste disposal activities and would not share services with other commercial endeavors (Bechtel 1979a).

The seaport would have facilities for receiving railway casks containing the waste canisters and for storing them in a water pool until shipment to the repository site. All required handling equipment, including that needed to load the canisters into seagoing vessels, would be available at the port.

The port facility could receive and handle 10,200 spent fuel canisters a year (Bechtel 1979b). For handling high-level reprocessing waste, the total annual throughput would be:

	<u>Canisters</u>
HLW	2,380
Cladding Hulls	2,300
End Fittings	<u>1,520</u>
Total	6,200

Cladding hulls and end fittings are not thermally hot. However, they would be handled in the same manner as HLW for storage and disposal because of their high radiation levels and the possibility of contamination by transuranic elements.

The shipping area of the port facilities would include a canister transfer pool and a transfer cask storage area. To load the ship, the canisters would be moved from the cask and transferred to the ship by crane. The dock facilities would accommodate two ships of the class described below.

b. Seagoing Vessels. Because of the quantities of waste canisters to be disposed of, subseabed disposal would require special dedicated ships (Bechtel 1979a). Each ship would contain equipment for handling the canisters during loading, a water pool to store the canisters during transportation, the necessary equipment to emplace the canisters in the sediment, and water cooling and treatment facilities.

The waste ships could have double hulls and bottoms. Waste canisters would be secured in the holds of the ships in basins filled with water. This concept of transporting fuel canisters in a shipboard storage pool, while new, is considered entirely feasible and is assumed for the reference study.

Disposal of spent fuel might require approximately 15 days to load a ship, 15 days for the round trip from port to repository, and up to 50 days to emplace the canisters at the subseabed site. Thus, a ship would make four trips a year. Based on transporting 1,275 canisters per trip, two ships would be required.

The sea-transportation requirements for HLW would be the same as those for spent fuel assemblies. It is estimated that the same numbers and class of ships as described above would be adequate for transporting HLW and cladding hulls. The same number of trips would be required, but total turnaround time would be about 15 days less because fewer canisters would be handled.

In addition to the ships used for the disposal operations, a survey ship would monitor the emplacement of canisters and their positions relative to one another.

Emplacement. It is assumed that a free-fall penetrometer would provide one alternative method for emplacing canisters in the seabed sediment (Bechtel 1979a). The canisters would have a nose cone to aid penetration and tail fins for guidance. Alternatively, they might be lowered to a predetermined depth and released, and would be designed to penetrate about 30 meters into the sediment. Laboratory tests indicate that the holes made as the canisters entered the sediment would close spontaneously. Canister instrumentation would permit a monitoring crew to track each canister to ensure proper penetration into the sediment and spacing between canisters.

The total seabed area required would be 560 km²/yr (215 mi²/yr) for HLW and 920 km²/yr (354 mi²/yr) for spent fuel assemblies, based on an arbitrary spacing of 300 m (984 ft) between canisters and a waste disposal system of 5,000 MTHM/yr.

Retrievability/Recoverability. Retrievability has not been designed into the system concept (though during the experimental period all emplaced radioactive material would be designed for retrievability) (DOE 1979c). Postemplacement waste-canister recovery from any of the four emplacement options (see Figure 6.1.12) would be possible with existing ocean engineering technology, but estimated costs are high.

Monitoring. After the wastes were emplaced, a monitoring ship would use instrumentation on the ship, on the ocean bottom, and on the canisters to determine information about the buried canister: e.g., its attitude and its temperature. This monitoring would continue for as long as necessary to verify the performance of the sub-seabed isolation system.

6.1.4.3 Status of Technical Development and R&D Needs

Present State of Development

The status of concept design, equipment, and facilities for different facets of a sub-seabed disposal operation is described below.

Emplacement Medium. Properties of the red clay sediment of the ocean's abyssal hills have been studied extensively under the Subseabed Disposal Program (SDP) (Talbert 1977, Sandia 1977, Sandia 1980). The considerable data collected indicate that the sediment is a very promising emplacement medium. The SDP has collected data on nuclide sorption and migration, effects of heat and temperature, ecosystems, and other aspects of the subseabed environment in these sediment areas. The program was started in 1973, and studies of the emplacement medium and of concept feasibility are planned to be completed in 1986. After that, the program would deal with other engineering problems, such as the handling of waste during sea transportation and emplacement (Sandia 1980).

Emplacement Methods. The SDP has not yet defined the methods of waste emplacement in the subseabed. The technical problems associated with this task would be addressed after the studies on sediment properties are completed. In other words, the required depth of emplacement, spacing of canisters, method for assuring hole closure, etc., would have to be known before emplacement methods could be developed.

Four possible methods of emplacement are being considered: (1) free-fall penetrometer, (2) winch-controlled penetrometer descent to a determined depth and final propulsion (the reference concept), (3) trenching, and (4) drilling. The operations are described in Reference 4. The first two methods that use penetrometers present fewer technical challenges since the penetrometer is a widely used tool in marine, land, space, and arctic operations.

Waste Form. The waste form and the canister design required for subseabed disposal of spent fuel have not been determined. Because of the high hydrostatic pressures at the ocean bottom, one important characteristic of the waste package would be a filler material with low compressibility. Generally, metallic fillers would satisfy this requirement, but other solid materials could be more acceptable because of cost advantages, resource conservation, and easier process technology.

The waste form required for storage of HLW in a subseabed repository has not been determined. It is believed that borosilicate glass might be adequate, especially if the temperature of the canister-sediment interface were maintained below 200 C (392 F). This would require adjusting the age of the waste and/or the diameter of the canister to provide rapid heat flow away from the canister. Other waste forms are also being considered.

Waste Containment. Due to the expected effects of high heat and radiation on the properties of the subseabed sediments, waste containment would have to be maintained for a few hundred years to delay the release of nuclides. Experimental data on the rate of corrosion of metallic materials in hot brine and seawater, collected primarily to improve the material performance in desalinization plants and in geothermal applications, would add to the confidence that this capability can be provided.

The SDP has also included laboratory experiments with metallic materials subjected to a seawater environment of 200 C (392 F) and 1,000 psi (6.9×10^6 Pa). Plates of Ticode 12 showed the lowest rate of corrosion, as determined by a weight-loss technique (Talbert 1979).

Facilities. The seaport storage facilities and the facilities that would have to be built aboard ship have not been developed. However, the technology for building them is available since they would resemble existing facilities, such as spent fuel storage pools and ordinary port facilities. The seaport location, size, and capabilities are not yet defined by the SDP.

Technical Issues

The engineering aspects for subseabed disposal have not been established. The transportation logistics, regulations, and the appropriate transportation "package" have not been developed. The precise size and type of facilities that would be built are not known, and the time and motion studies to select the optimum ship size have not been made. In addition, a large area of uncertainty revolves around the methodology that would be used to emplace the waste. Techniques to ensure that waste canisters were placed deep enough into the sediment have not been demonstrated.

If demonstrated, a major attribute of subseabed disposal would be the ability of the sediments to hold radionuclides until they had decayed to innocuous levels. To determine whether these sediments could actually do this, the following technical issues would need resolution.

Ion Transport in the Sediment. More data would be required regarding the rates at which the radioactive ions transfer through the sediment. Studies and empirical data would be required to determine the thermal interaction with canister materials and wastes, conduction, and convection through the sediment.

Ion Transport to the Biosphere. The paths and rates at which the radioactive ions could transfer from the sediment, through the benthic boundary layer, and into the water column are not known. Both mathematical models and empirical experiments would be required to obtain this information. Modeling would also be required to determine a realistic rate of migration up the water column.

Sediment Mechanical Requirements. The subseabed sediments that would be candidates for nuclear waste disposal are between 4,000 and 6,000 m (13,000 and 20,000 ft) below the ocean surface. Further information would have to be acquired regarding their macroscopic (as well as microscopic) structural characteristics. These characteristics include sediment closure after emplacement and long-term sediment deformation and buoyancy resulting from heating.

R&D Requirements

The SDP is divided into seven R&D fields of study (see Sandia 1980), each with numerous subdivisions. As far as funding and the state of technology allow, all of these studies are being pursued simultaneously, though not all at the same level of detail. An eighth field, safeguards and security, would be established later as the results of the other seven studies become known. Brief descriptions of these eight studies which define R&D requirements, follow:

Site Studies. Current studies include evaluation of North Atlantic and North Pacific oceanic areas that meet site suitability criteria. From these areas, certain study locations have been, and will continue to be, identified for more intensified study.

Environmental Studies. Environmental studies include physical and biological oceanography. They focus on analyzing physical characteristics of the water column from the ocean surface to the sediment surface, and on gathering all pertinent information about the marine life that inhabits the water column. The ultimate purpose of these studies is to determine whether, and to what degree, the physical and biological characteristics of the ocean would accelerate or slow the transport of accidentally released radionuclides to man's environment.

Multibarrier Quantification. The multibarrier study includes the sediment, the canister, and the waste form, both immediately adjacent to the waste container and further afield, to determine their natural characteristics. Again, the ultimate purpose is to learn whether, and to what degree, they would allow released radionuclides to be transported. A second purpose is to learn how they would react to the heat and radiation generated by a waste container, as well as to any engineered modification to the sediment such as artificial closing of the emplacement hole.

Transportation. Transportation studies include four subdivisions:

- Land transport with investigations directed to transporting HLW and/or spent fuel from an originating plant to the port facility by rail, road, or barge.
- The port facility, including a receiving structure.
- The staging area, to include cooling facilities for holding waste packages until they could be loaded.
- Sea transport with studies including design of special transport/emplacement vessels and of travel routes designed to minimize interaction with shipping lanes and all other forms of maritime activity. It is likely that this would be a self-powered ship, but it could be a vessel that could be towed, possibly under water. Transportation technology is in early planning stages, pending determination of disposal feasibility.

Emplacement and Monitoring. The study of emplacement and monitoring focuses on the time period that begins when waste packages would be removed from their cooling area on the transport vessel and continues through burial deep in the subocean sediments and closure of the entrance hole, either naturally or artificially. An intrinsic part of the process would be the monitoring function. Monitoring would include surveying precise disposal locations, guiding emplacement mechanisms into those locations, and tracking the integrity, attitude, and stability of waste containers for as long as would be required after emplacement.

Social/Political Studies. Even if technological and environmental feasibility for the subseabed disposal concept were established, domestic and international institutions would ultimately determine whether the concept could be used. There are no laws or agreements at this time that specifically prohibit or allow subseabed disposal. Issues important to this area are further discussed in Section 6.1.4.4 under International and Domestic Legal and Institutional Considerations. International agreements and structures would enhance the implementation of the concept. Evaluation of the current political and legal postures of all countries that might be involved in subseabed disposal is under way. The existence of an international NEA/OECD Seabed Working Group is indicative of the international interest in the concept.

Risk/Safety Analyses. As data become available, risk and safety analyses would be completed on all aspects of the SDP.

Security and Safeguards. Except in the most general terms, studies in these areas would have to await data acquisition and assessment.

R&D Costs/Implementation Time

Research and development is assumed to end when the technology had been translated into routine practice at the first facility. Follow-on R&D in support of facility operation is considered in a different category.

To date, almost all resource expenditures have been focused on the technical and environmental feasibility of the subseabed geologic concept, rather than on specific on-site studies or demonstrations of current engineering practice. The estimated total R&D costs are \$250 million (DOE, 1979).

The SDP program plan has been divided into four distinct phases (Sandia, 1980). In each phase, the concept feasibility is assessed. The estimated completion dates shown do not consider programmatic perturbances resulting from regulatory or institutional influences.

- Phase 1 Estimation of technical and environmental feasibility on the basis of historical data. Completed in 1976.
- Phase 2 Determination of technical and environmental feasibility from newly acquired oceanographic and effects data. Estimated completion date: 1986.
- Phase 3 Determination of engineering feasibility and legal and political acceptability. Estimated completion date: 1993-95.
- Phase 4 Demonstration of disposal facilities. Estimated completion date: 2000 to 2010 (Anderson et al. 1980).

Summary

Major uncertainties, shortcomings, and advantages of the concept are summarized below:

- The remoteness of the location, apparent sorption capacity of the sediments, and demonstrated stability of the site are attractive attributes.
- The concept could be implemented in a step-wise fashion.
- The expected performance of packages and waste form in the environment at the seabed is not well understood.
- Specific new domestic legislation and international agreement would likely be required.
- Retrievability to allow for corrective action purposes might be difficult.
- Transportation requirements to a remote location add to the overall risk of the concept.

6.1.4.4 Impacts of Construction and Operation (Preplacement)

Health Impacts

Both radiological and nonradiological health impacts are discussed below.

Radiological Impacts. Both occupational and nonoccupational doses prior to the waste arriving at the seaport facility are expected to be similar to those anticipated for a mined geologic repository, as presented in Chapters 4 and 5.

The occupational and nonoccupational radiological impacts of the operation of the seaport facility and the seagoing vessels have been developed by Bechtel (1979a), and are presented in Table 6.1.11. These impacts are conservatively estimated as equivalent to those for away-from-reactor storage pools (AFR), corrected in consideration that:

- The primary waste handled at the subseabed facilities would be 10 years old.
- The primary waste at the subseabed facilities would be encapsulated.
- The number of personnel is expected to be smaller at the seaport facility than at the AFR facility. This may be offset by the fact that personnel might receive occupational doses for longer time periods while serving aboard ship.

TABLE 6.1.11. Radiological Impacts Of The Normal Operation
At A Subseabed Repository

	Whole Body Dose, man-rem/yr	
	Spent Fuel	High-Level Waste
Occupational		
Seaport Facility	340	200
Seagoing Vessels	340	200
Nonoccupational		
Seaport Facility	40	10
Seagoing Vessels	Negligible	Negligible

Bechtel (1979a) gives the consequences of abnormal events at subseabed facilities. These consequences are equated with accidents postulated for the AFR (i.e., design basis tornado) facility for the most exposed public individual. No probability analysis was included. For spent fuel disposal, the radiological impacts of an abnormal event would be 0.02 mrem/event for the seaport facility and 0.003 mrem/event for the seagoing vessels. For HLW, these impacts would be 0.001 mrem/event and 0.002 mrem/event, respectively.

The maximum risk would be posed by the sinking of the seagoing vessel or by loss of waste canisters overboard. Except for accidents in coastal waters where mitigation actions could be taken, the radioactive materials released into the sea following such an event would disperse into a large volume of the ocean. Some radionuclides might be reconcentrated through the food chain to fish and invertebrates, which could be eaten by man. Bechtel (1979a) assumes that the waste could be retrieved if either event were to occur and does not provide an impact estimate. The doses provided in Table 6.1.12 for such an event are taken from EPA (1979).

Nonradiological Impacts. The numbers of injuries, illnesses, and deaths related to the construction and operation of the subseabed disposal option prior to the waste arriving at the seaport facility/repository are expected to be similar to those for the mined geologic options. At the seaport facility, it is estimated that the impacts would be no greater than those associated with surface storage and transfer facilities to be used with a reprocessing plant or spent fuel overpacking facility. These impacts are discussed in Chapter 4.

Additional areas specific to subseabed disposal that would have nonradiological health impacts are the construction of seagoing vessels and the conduct of operations at a seaport and on the ocean. Although there are no quantitative estimates of these impacts, it is anticipated that they would be similar to those incurred during the construction and operation of conventional seagoing vessels and operation of conventional dock facilities.

Natural System Impacts

Impacts to the natural environment for this disposal option would be related primarily to transportation and emplacement activities. Radiological concerns would be most significant

TABLE 6.1.12. Estimated Dose Commitment From Marine Food Chain For Loss of Waste At Sea

	<u>Population</u> <u>man-rem</u>	<u>Average Individual,</u> <u>rem</u>
Undamaged Spent Fuel		
Continental Shelf	510	5.9×10^{-4}
Deep Ocean	100	1.1×10^{-4}
Damaged Spent Fuel		
Continental Shelf	1×10^5	0.11
Deep Ocean	100	1.1×10^{-4}
HLW (Plutonium Package)		
Continental Shelf	Not provided	Not provided
Deep Ocean	100	1.1×10^{-4}

under abnormal conditions, while nonradiological impacts could also pose problems under normal operating conditions.

Transportation-related impacts for those activities occurring before the waste material was loaded on the ships would be similar to those for a mined geologic repository. Once the material was loaded onto the ships, impacts to the marine environment would have to be considered. In the case of potential accident conditions at sea, the design of the waste transporting vessels to include double hulls and bottoms would reduce the likelihood of releasing harmful material into the environment.

There are several uncertainties that limit the ability to predict natural system impact levels with confidence. Of primary concern is a lack of understanding of ion transport within the sediment and biosphere, including the benthic region, the water column and ocean life forms. In addition, the extent of the isolation barrier that the resealed sediment would provide after emplacement is not clear. Each of these factors makes detailed impact assessment difficult.

Other subseabed disposal impacts identified, but not quantified by Bechtel (1979a), include minor air emissions from construction equipment, dust generation, and road, rail, and vessel emissions. Construction-related impacts on water quality and vegetation as well as impacts on the marine environment resulting from dredging and breakwater construction could be locally significant. Although these impacts were identified by Bechtel (1979a), there are no data that indicate they would be significant.

Socioeconomic Impacts

Because a major land repository would not be required under this option, the most important socioeconomic impacts would be attributable to transportation activities. Transportation activities fall into three categories: (1) transportation of wastes on land to the port where the wastes would be transferred to the ship, (2) waste-handling activities at the port facility, and (3) ocean transportation from the port facility to the point where the material would be deposited in the seabed sediment.

Socioeconomic impacts would be concentrated at the point where support activities were most intense: at the port facility. The nature of the activity has led certain reviewers to conclude that one of the most significant factors associated with this disposal option would be difficulty in finding a suitable dedicated (Bechtel 1979a). Moreover, they project moderate community impacts and suggest that local socioeconomic impacts could reach significant levels.

Detailed projections of the impact of implementing this disposal option on the public and private sectors could be made only on site-specific basis. Nevertheless, impacts would be expected in the coastal area near the port facility. The total anticipated increase in employment for a 5000 MTHM per year disposal system, although quite concentrated, is expected to be less than 2000 people.

Aesthetic Impacts

The significance of aesthetic impacts would depend on the appearance and operating parameters of a facility, as well as on the extent to which it would be perceived by humans. For the subseabed disposal option, much of the waste-handling and transportation activities would occur in remote areas of the ocean. Consequently, the aesthetic impacts, regardless of their nature, would not be significant.

Aesthetic impacts near the port facility, however, could be locally significant. Such impacts could be accurately determined only on a site-specific basis. However, it is important to recognize that the required port facilities for a nuclear waste handling facility would be substantial.

Resource Consumption

Use of energy and construction of seaport facilities and seagoing vessels would be the primary resource consuming activities in this option. Energy would be consumed during land transportation, loading, and sea transportation activities. A quantitative estimate of energy consumption is provided in Table 6.1.13.

The seaports would have facilities for receiving railway casks containing the waste canisters and for placing them in interim storage. Interim storage pools should be able to handle one-half of the anticipated yearly volume of wastes (2500 MTHM) and are expected to

TABLE 6.1.13. Estimated Energy Consumption

	Spent Fuel	HLW
Propane m ³	2.4 x 10 ⁴	1.0 x 10 ⁷
Diesel, m ³	5.0 x 10 ⁶	1.6 x 10 ⁶
Electricity, KWh	2.0 x 10 ¹⁰	5.7 x 10 ¹⁰

require an area within the boundaries of the port area subseabed support facilities of 2320 m² (25,000 ft²) (Bechtel 1979a). Other storage and transfer facilities would also be needed. The total area required for all the required facilities is expected to be over 3600 ha (8500 acres).

Construction of the waste disposal ships with double hulls and bottoms, waste handling equipment for loading, and carefully constructed compartments for holding the wastes during transportation activities, like construction of the port facilities, would lead to the consumption of steel and other basic construction materials. An estimate of the material consumption is provided in Table 6.1.14.

International and Domestic Legal and Institutional Considerations

The subseabed disposal option, like the island and ice sheet options, would require transporting waste material over the ocean, and the general international implications of such transportation are important.

Any implementation of subseabed disposal is far enough in the future that many current legal and political trends could change. However, it is not too early to identify important problems, so that possible developments could be foreseen and controlled.

The use of subseabed disposal would be governed by a complex network of legal jurisdictions and activities on both national and international levels. Domestic use of subseabed disposal of radioactive waste would require amendment of the U.S. Marine Protection, Research, and Sanctuaries Act of 1972 (The Ocean Dumping Act) which currently precludes issuance of a permit for ocean dumping of high-level radioactive waste.

Table 6.1.14. Estimated Material Consumption for Ship and Facility Construction (in MT)

	Spent Fuel	HLW
Carbon Steel	877,000	282,000
Stainless Steel	83,500	22,500
Components		
Chromium	14,200	4,600
Nickel	7,500	2,000
Tungsten	---	---
Copper	1,400	1,900
Lead	12,900	2,900
Zinc	1,200	600
Aluminum	13,000	1,400

The London Convention of 1972, a multinational treaty on ocean disposal, addresses the problem of dumping of low-level and TRU wastes at sea and bans the sea dumping of high-level

wastes (Deese 1976). This treaty is currently being revised to deal more specifically and completely with the problem of dumping low-level and some TRU wastes. This treaty arguably does not preclude the controlled emplacement of high-level wastes or spent fuel into geologic formations beneath the ocean floor. However, the intended prohibition of the treaty would require clarification.

Subseabed disposal might offer the important political advantage of not directly impacting any nation, state, or locality. Likewise, the alternative might have the disadvantage of incurring risk to nations that do not realize the benefits of nuclear power generation.

Assuming that the real impact uncertainties associated with the subseabed concept were resolved, the primary political disadvantage of subseabed disposal would be its possible perception as an ecological threat to the oceans. If publics, governments, and international agencies were to view such disposal as merely an extension of past ocean dumping practices, implementation would be difficult if not impossible. However, if this option were understood as involving disposal in submarine geologic formations that have protective capacities comparable to or greater than similar formations on land, opposition might be less.

6.1.4.5 Potential Impacts Over the Long Term (Postemplacement)

Potential Events

Earthquakes, volcanic action, major climatological and circulatory changes, and meteorite impacts are examples of natural processes that might affect subseabed containment stability. Careful selection of the ocean area would minimize the probability of the first three events occurring. There is no known method of minimizing the probability of meteorite impact other than concentrating emplacement, which, while reducing the random target area, would correspondingly increase the potential consequences if a meteorite did strike. On the other hand, other damage caused by any meteorite that could penetrate 5 km (3 mi) of water would make the release of emplaced radioactive waste insignificant.

For HLW disposed of in a subseabed repository, a very low probability for criticality is assumed because of the great distances between canisters at the bottom of the sea. For spent fuel, the probability of criticality might be somewhat greater because of the higher fissile content of a single canister.

Since the site would be located in a part of the ocean with no known materials of value, future human penetration would be highly unlikely.

Potential Impacts

Two models have been developed by Grimwood and Webb (1976) to characterize the physical transport and mixing processes in the ocean, as well as incorporation in marine

food chains and ultimate consumption of seafood and radiation exposures to man. Although there is some question as to the applicability of these models to the subseabed disposal option, the following summary of results using these models is presented until such time as better estimates of radiation exposures to man from subseabed disposal are available.

The individual doses resulting from the consumption of surface fish, deep-ocean fish, or plankton are expected to be well below the maximum permissible levels. External individual doses^(a) from contamination of coastal sediments are expected to be fractions of the ICRP dose limit for both skin and whole body irradiation. The largest annual internal population doses to the whole body and bone due to the consumption of surface fish would be about 4×10^4 and 10^5 man-rems, respectively. The largest annual external population doses from contaminated sediments would be about 10^3 to 10^4 man-rems for both skin and whole body. These large population doses would occur during the early stages of postemplacement and would decrease during the later stages.

As an attempt to provide a further yardstick against which to compare the results of the calculations, Table 6.1.15 gives the concentrations of nuclides predicted by the modeling, as well as the natural activity in seawater.

6.1.4.6 Cost Analysis

An estimate of capital, operating, and decommissioning costs for subseabed disposal has been made for both spent fuel disposal and HLW disposal (Bechtel 1979a). Both are based on penetrometer emplacement. All estimated costs are in January 1978 dollars.

TABLE 6.1.15. Levels Of Natural And Wastes Radionuclides In Seawater

Nuclide	Natural Activity In Seawater, Ci/cm ³	Max Widespread Surface Water Conc. Predicted From Postulated Waste Disposal Operation, Ci/cm ³ (No Containment)
Actinides		
Pb-210	(1 - 9) $\times 10^{-11}$	2×10^{-15}
Pb-210	1×10^{-10}	2×10^{-15}
Ra-226	1×10^{-10}	2×10^{-15}
Th-230	(0.6 - 14) $\times 10^{-13}$	2×10^{-17}
Th-234	1×10^{-9}	1×10^{-15}
U-234	1×10^{-9}	1×10^{-15}
U-238	1×10^{-9}	4×10^{-15}
Pu-239		1×10^{-12}
Fission Products		
H-3	2×10^{-10}	1×10^{-12}
Sr-90		4×10^{-10}
I-129	3×10^{-11}	3×10^{-14}
Cs-137		6×10^{-10}

(a) Based on world population

In each case, only those costs associated with and peculiar to subseabed disposal are addressed. Facilities common to all disposal options under consideration, such as transportation and geologic repository facilities, are not specifically addressed.

Capital Costs

The capital costs for the subseabed disposal alternative are categorized as follows.

Seaport Interim Storage Facility. This installation would provide receiving facilities for 5,000 MTHM/yr of spent fuel assemblies in 10,200 canisters. It would also be designed to provide interim storage for 5,000 canisters (2,500 MTHM). The same facility would receive the HLW and hulls from a 5,000 MTHM/yr fuel recycling system. Interim storage would be provided for 3,100 of these canisters at the port facility.

The seaport interim storage facility would be similar to a packaged fuel receiving and interim storage facility (Bechtel 1977) appropriately adjusted for size and waste form. The capital cost estimates are \$240 million for the spent fuel case and \$190 million for the HLW case.

Port Facility. The port facilities for both disposal cases are assumed to be identical for cost estimating purposes. The capital cost estimate is based on a recent estimate of another facility (Bechtel 1979a). The estimate for this port is \$24 million.

Disposal Ships. The two disposal ships for the spent fuel case would have a capacity of 1,275 canisters each, while those for the HLW case would have a capacity of 775 canisters each. Since the canister capacity difference would be offset by the heat load and cooling requirement difference, the ships are assumed to be identical for estimating purposes.

The capital cost estimate of the ships is based on an estimate for a mining ship (Global Marine Development, Inc. 1979) appropriately adjusted. The estimated capital cost of the two disposal ships is \$310 million (\$155 million each). Note however that sophisticated offshore oil well drilling ships have been reported to cost between \$50 million and \$70 million each (Compass Publications 1980) or about half the above estimate.

Monitoring Ship. The capital cost for the monitoring ship was estimated from available data for oceanographic vessels. The estimate is \$3.0 million for the ship and an additional \$0.9 million for navigation and control, special electronics, and other surveillance equipment and for owner's costs. This brings the total capital cost to \$3.9 million (Treadwell and Keller 1978).

Operating Costs

Operating costs for the subseabed disposal concept are estimated on a per year basis based on 5,000 MTHM/yr of both waste forms (spent fuel and HLW). This would result in virtually the same sea transportation requirements (number of trips per year). However, differences would occur for the HLW disposal case in years 1 through 9, when only hulls would be

processed and disposed of, and during years 41 through 49, when only HLW would be disposed of.

The estimated yearly operating costs for the subseabed disposal concepts are presented in Table 6.1.16.

Operating costs associated with the reference subseabed disposal concept but also common to other disposal concepts are assumed to be similar. These costs would include transportation, AFR facilities (for the spent fuel), P/E facilities, and geologic repository facilities (assumed for the reference concept).

Decommissioning Costs

Decommissioning costs particularly associated with subseabed waste disposal operations would probably be limited to the seaport, interim storage facility, the port facility, and the disposal ships. The monitoring ship is not expected to be affected by radioactive waste during its 40 years of operation. Any decommissioning costs associated with the monitoring ship are assumed to be offset by its salvage value, which results in a zero net decommissioning cost.

The decommissioning cost of an AFR facility is used as the basis for the decommissioning cost of the seaport interim storage facility (Bechtel 1979b). These costs, based on 10 percent of capital cost excluding owner's cost, are approximately \$23 million for the spent fuel disposal and approximately \$18 million for the HLW disposal case.

The decommissioning costs for the port facility and two disposal ships are the same for both waste forms and are estimated to be about \$2 million and \$29 million, respectively, assuming 10 percent of capital cost less owner's costs.

Costs for decommissioning other facilities associated with subseabed disposal and common to other waste disposal alternatives are assumed to be similar. These facilities include AFR facilities (for the spent fuel), P/E facilities, and geologic repository facilities. These

TABLE 6.1.16. Estimated Operating Costs

Facility	Estimated Cost, \$ million/yr	
	Spent Fuel Disposal	HLW Disposal
Seaport Interim Storage Facility		
Years 1-9	---	3.4
Years 10-40	6.2	4.9
Years 41-49	6.2	3.4
Port Facility	1.5	1.5
Disposal and Monitoring Ships		
Years 1-9	---	14.5
Years 10-40	20.9	20.9
Years 41-49	20.9	14.3

total costs are estimated to be about \$398 million for the spent fuel disposal and \$721 million for the HLW disposal.

6.1.4.7 Safeguard Requirements

Because this concept may involve both subseabed and mined geologic disposal, its implementation could require safeguarding two separate disposal paths. The risk of diversion for the subseabed disposal concept would be primarily a short-term concern because of the remoteness of the disposal site and the major operational and equipment requirements that would have to be satisfied for retrieval. Physical protection of the sensitive facilities and transportation operations would be the most effective way to deny access to the waste for the short term, as is common to most waste disposal concepts. See Section 4.10 for additional discussion of predisposal operations safeguards requirements.

6.1.5 Ice Sheet Disposal

6.1.5.1 Concept Summary

It is estimated that, without significant climatic changes, the continental ice sheets could provide adequate isolation of high-level radioactive waste from the earth's biosphere. However, the long-term containment capabilities of ice sheets are uncertain. Areas of uncertainty have been reviewed by glaciologists (Philberth 1958, Zeller et al. 1973, and Philberth 1975). These reviewers cited the advantages of disposal in a cold, remote, internationally held area and in a medium that should isolate the wastes from man for many thousands of years to permit decay of the radioactive components. But they concluded that, before ice sheets can be considered for waste disposal applications, further investigation is needed on:

- Evolutionary processes in ice sheets
- Impact of future climatic changes on the stability and size of ice sheets.

Most of the analysis in these studies specifically addresses the emplacement of waste in either Antarctica or the Greenland ice cap. Neither site is currently available for waste disposal for U.S. programs: Antarctica because of international treaties and Greenland because it is Danish territory.

Proposals for ice sheet disposal suggest three emplacement concepts:

- Meltdown - emplaced in a shallow hole, the waste canister would melt its own way to the bottom of the ice sheet
- Anchored emplacement - similar to meltdown, but an anchored cable would allow retrieval of the canister
- Surface storage - storage facility would be supported above the ice sheet surface with eventual slow melting into the sheet.

Ice sheet disposal, regardless of the emplacement concept, would have the advantages of remoteness, low temperatures, and isolating effects of the ice. On the other hand, transportation and operational costs would be high, ice dynamics are uncertain, and adverse global climatic effects are a possibility.

6.1.5.2 System and Facility Description

Systems Options

The reference concept for the initial ice sheet disposal of nuclear waste has been developed from a number of options available at each step from the reactor to disposal in the ice sheet. It includes the three basic emplacement options and was selected through judgment of a "most likely" approach based on available information and is not supported by a detailed system engineering analysis.

Various options to be considered for ice sheet disposal are summarized in Figure 6.1.14. The bases for selection of the options chosen for the reference design (those blocked off) are detailed in a variety of source material cited in Appendix M.

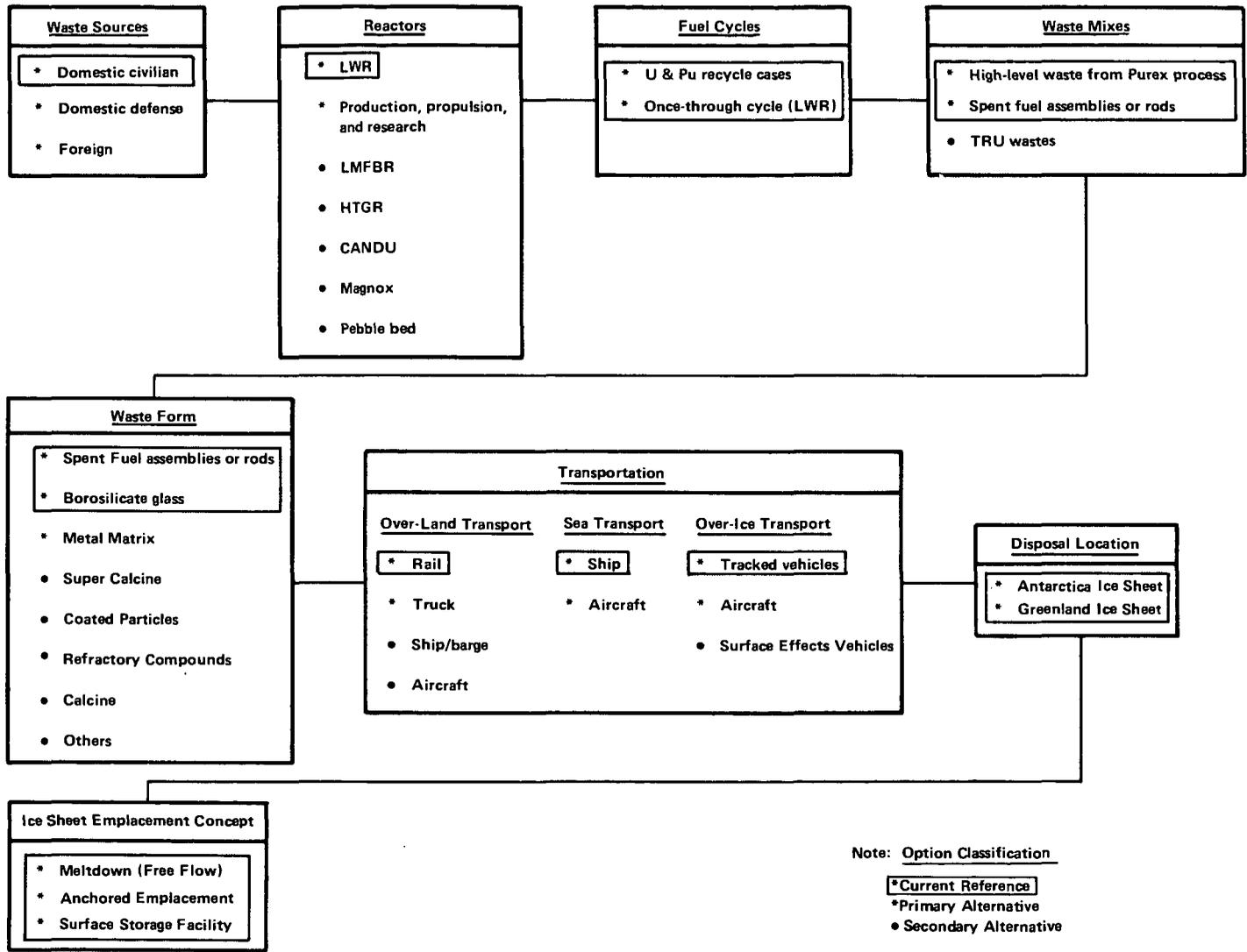


FIGURE 6.1.14. Major Options for Ice Sheet Disposal of Nuclear Waste

Because the options for the waste disposal steps from the reactor up to, but not including, the transportation alternatives are similar to those for a deep geologic repository, the options selected for the reference design are similar for the two concepts. From that point on, the options selected for the reference ice sheet design are based on current program documentation for ice sheet disposal.

Waste-Type Compatability

Ice sheet disposal by meltdown has been considered primarily for solidified, high-level wastes from nuclear fuels reprocessing. It would also be applicable for direct disposal of spent fuel, without reprocessing, although meltdown would be marginal if the fuel were emplaced 2 years after reactor discharge. The feasibility of meltdown emplacement of cladding hulls and fuel assembly hardware is questionable because the canister heating rate from radioactive decay would be less than 1/10 that in HLW waste canisters.

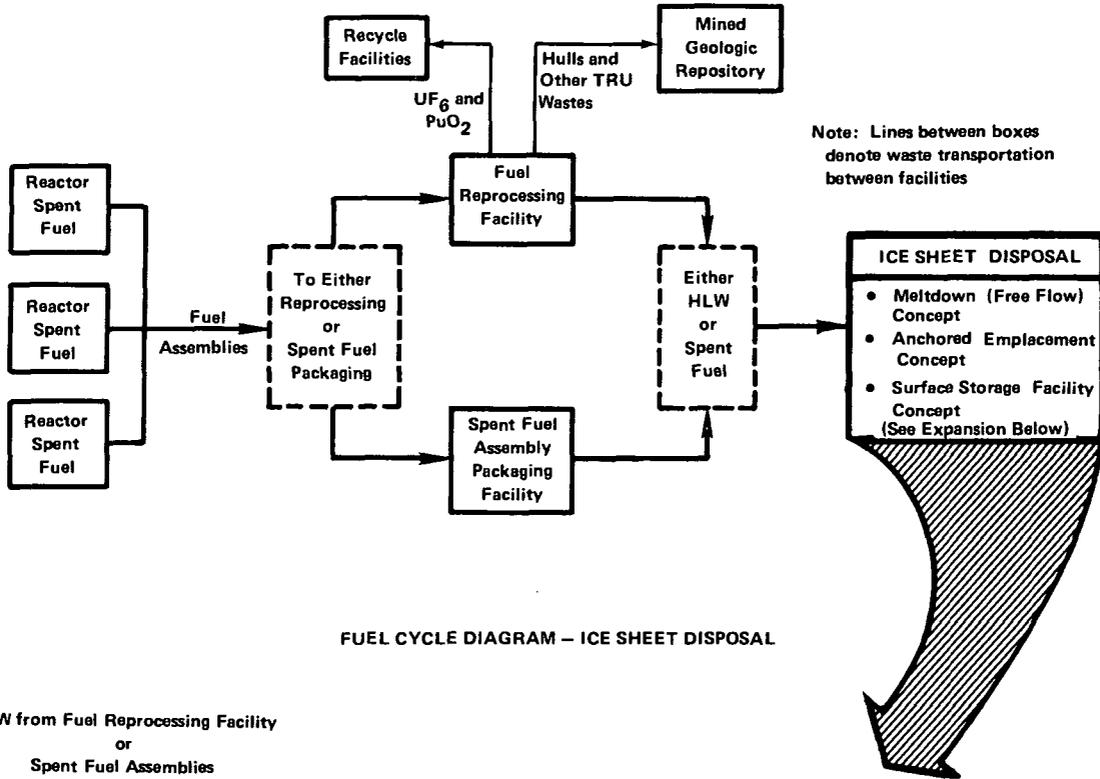
For most TRU waste, the heating rate would be less than 1/1000 that expected in HLW waste canisters, and the meltdown concept does not appear to be feasible. Without blending with HLW, disposal of this waste would be limited to storage in surface facilities on the ice or emplacement in shallow holes in the ice. For these options, the waste would be buried gradually in the ice sheet. Contact handled and remotely handled TRU wastes could be handled in a similar manner. Because of volume and cost considerations, TRU wastes are assumed to be placed in other terrestrial repositories.

Waste System Description

The ice sheet waste management system is detailed in Figure 6.1.15. This system concept is very similar to the very deep hole concept since both spent fuel and the uranium-plutonium recycle cases could be treated and mined geologic repositories could augment disposal.

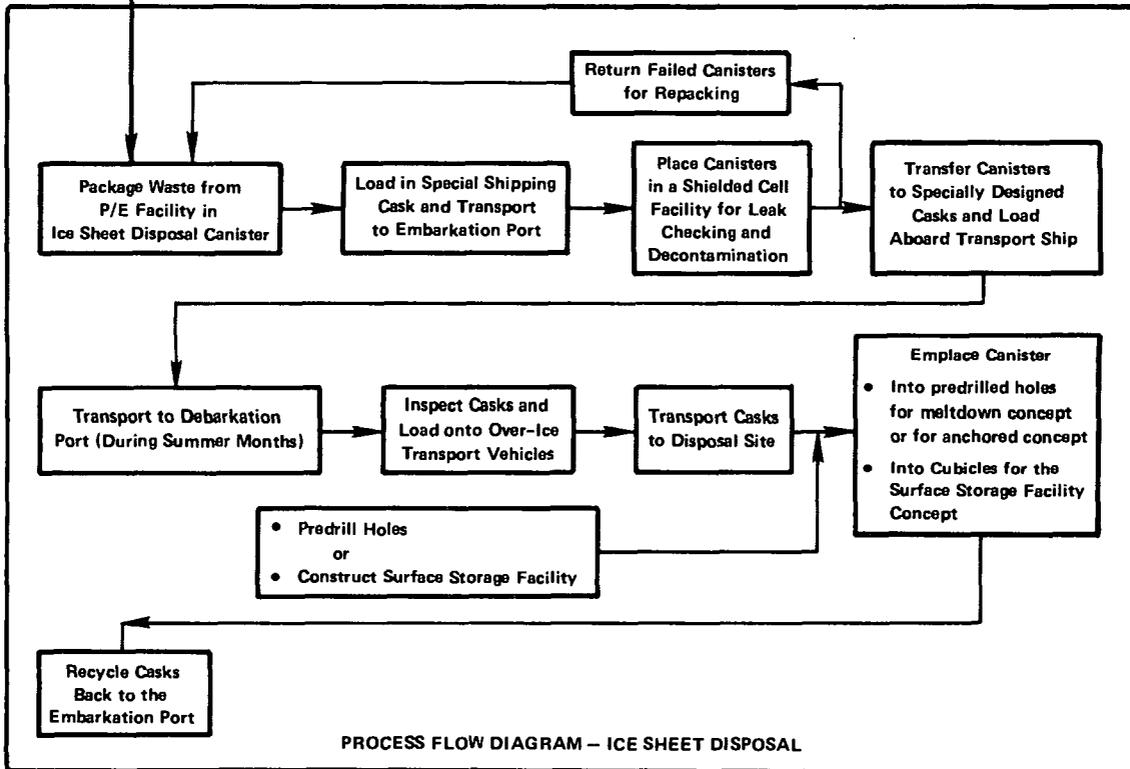
The reference ice sheet disposal concept is not yet well defined. None of the three basic emplacement concept alternatives proposed in the literature (Battelle 1974, EPA 1979, and ERDA 1976) has been selected as a reference or preferred alternative. Waste disposal by any one of these three concepts would be either in the Antarctica or Greenland ice sheets. A generalized schematic of the waste management operational requirements is provided in Figure 6.1.16 (Battelle 1974). The schematic shows the basic system operations (EPA 1979):

- Predisposal treatment and packaging at the reprocessing plant
- Transporting solidified waste from the reprocessing plant or interim retrievable surface storage facility by truck, rail, or barge to embarkation ports
- Marine transport by specially designed ships during 1 to 3-month periods of each year.
- Unloading the waste canisters at a debarkation facility near the edge of the land mass
- Transporting over ice by special surface vehicles or aircraft on a year-round basis, as practicable
- Unloading and emplacing the waste canisters at the disposal site.



FUEL CYCLE DIAGRAM - ICE SHEET DISPOSAL

HLW from Fuel Reprocessing Facility
or
Spent Fuel Assemblies



PROCESS FLOW DIAGRAM - ICE SHEET DISPOSAL

FIGURE 6.1.15. Waste Management System--Ice Sheet Disposal

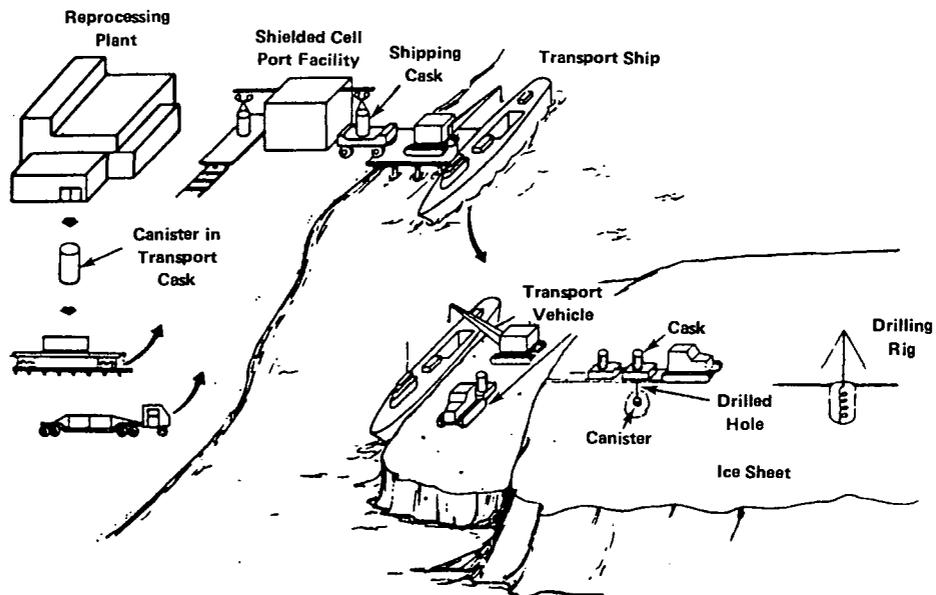


FIGURE 6.1.16. Schematic of Operations in Ice Sheet Disposal Systems for High-Level Radioactive Wastes (18)

Predisposal Treatment and Packaging. The predisposal treatment of waste for the ice sheet concept would be identical in many respects to the predisposal treatment of waste for the mined geologic repository concept. Chapter 4 discusses the predisposal systems for both spent fuel and HLW common to all the various alternative concepts for waste disposal.

Transportation and Handling. Transportation to the disposal site would probably be accomplished in three steps, as indicated above. First, all the waste canisters would be loaded into heavily shielded transport casks for shipment from the interim storage site to the embarkation port. Waste containers would accumulate at the embarkation port in the U.S. on a year-round schedule. There, the canisters would be unloaded in a shielded cell facility and examined for leakage, contamination, damage, or other unsuitable conditions. The canisters would be overpacked, transferred individually to specially designed casks, and loaded aboard a specially designed transport ship for shipment to the ice sheet. Acceptable canisters could also be stored for up to a year in an interim retrievable surface storage facility (Szulinski 1973). Any unacceptable canister would either be corrected on site or returned to the reprocessing plant or another appropriate handling facility.

Landing and discharge operations at the ice sheet would require special facilities and would be limited to the summer months. At the debarkation port, the casks would be inspected and unloaded onto over-ice transport vehicles. After transport to the disposal site, the canisters would be lowered from the casks to the emplacement site and the casks would be recycled back to the embarkation port. An alternative transportation mode would be to fly the waste canisters from the debarkation site to the emplacement site.

It appears possible, as an alternative, that the same shipping cask might be used for handling a waste canister first at the reprocessing plant, then for marine transport to the ice sheet, and finally for over-ice transport to the disposal site.

Debarkation ports on the ice sheets with handling systems for unloading casks directly onto the over-ice transport system would be possible in the Antarctic or in Greenland, but might be very expensive. The currently preferred alternative is to dock the transport ship at a land-based port in an ice-free area to unload the casks into the over-ice transport vehicles.

Emplacement. The waste canisters would be disposed of using one of the three basic concepts described in detail below.

The melt-down or free flow concept is shown in Figure 6.1.17 (ERDA 1976). Waste would be disposed of by selecting a suitable location in the ice sheets, predrilling a shallow hole, lowering the canister into the hole, and allowing it to melt down or free flow to the ice sheet base and bedrock beneath (EPA 1979).

The surface holes would be predrilled to depths from 50 to 100m and would provide protective shielding from radiation during canister emplacement. To avoid individual canisters interfering with each other during descent and possible concentration at the ice sheet base, the suggested spacing between holes is about 1000 m.

The canister meltdown rate is based on calculations from the penetration rates of thermal ice probes. It is estimated that the rate of descent for each canister would be on the order of 1.0 to 1.5 m/day. Assuming only vertical movement and an ice sheet 3000 m (9900 ft) thick, meltdown to the bedrock would take 5 to 10 years.

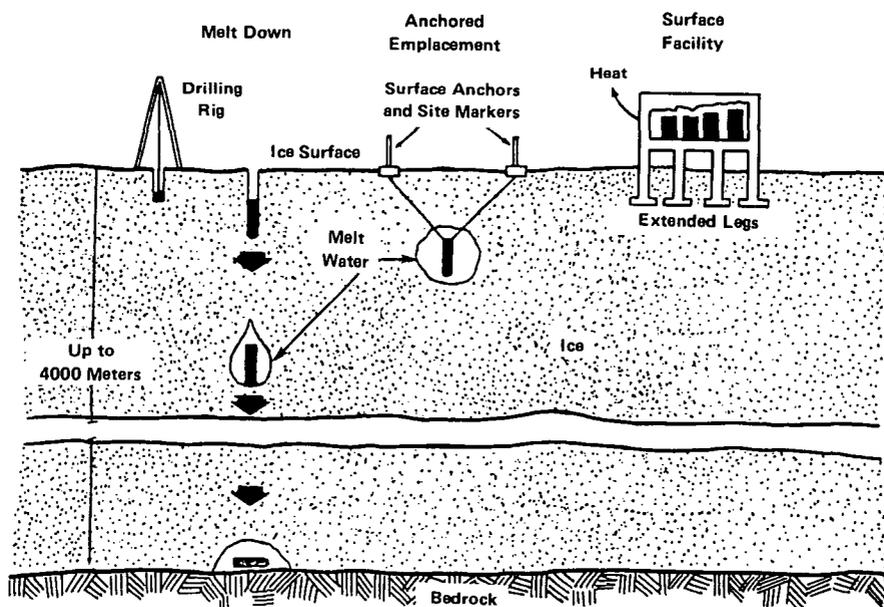


FIGURE 6.1.17. Ice Sheet Emplacement Concepts

An important factor in this concept would be the design and shape of the canister, which should help assure a vertical path from surface to bedrock. In addition to the canister design and shape, the type of construction materials would be important. Specifications for these materials would have to include consideration of differences in ice sheet pressure and the possibility of saline water at the ice/ground interface. A multibarrier approach that gives consideration to the total waste package and its emplacement environment would be required. This approach would be equally applicable to the anchored emplacement and surface storage alternatives.

The anchored emplacement concept, also shown in Figure 6.1.17, would require technology similar to that required by the meltdown or free flow concept described above, the difference being that this concept would allow for interim retrieval of the waste (EPA 1979). Here, cables 200 to 500 m (660 to 1650 ft) long would be attached to the canister before lowering it into the ice sheet. After emplacement the canister would be anchored at a depth corresponding to cable length by anchor plates on or near the surface. The advantage over the meltdown concept is that instrument leads attached to the lead cable could be used to monitor the condition of the canister after emplacement.

Following emplacement, new snow and ice accumulating on the surface would eventually cover the anchor markers and present difficulties in recovery of the canister. The average height of snow and ice accumulating in the Antarctic and Greenland is about 5 to 10 cm/yr (2 to 4 in./yr) and 20 cm/yr (8 in./yr), respectively. However, climatic changes might result in a reversal of this accumulation with ice being removed from the surface by erosion or sublimation. If continued for a long period of time such ice surface losses could expose the wastes. Recovery of canisters 200 to 400 years after emplacement might be possible by using 20-m (66-ft)-high anchor markers. It would take about 30,000 years for the entire system to reach ice/ground interface at a typical site. During that time, the canisters and anchors would tend to follow the flow pattern of the ice (Battelle 1974).

The surface storage facility concept would require the use of large storage units constructed above the snow surface (EPA 1979). The facilities would be supported by jack-up pilings or piers resting on load-bearing plates, as shown in Figure 6.1.17. The waste canisters would be placed in cubicles inside the facility and cooled by natural draft air. The facility would be elevated above the ice surface for as long as possible to reduce snow drifting and heat dissipation. During this period, the waste canisters would be retrievable. However, when the limit of the jack-up pilings was reached, the entire facility would act as a heat source and begin to melt down through the ice sheet. It is estimated that such a facility could be maintained above the ice for a maximum of 400 years after construction (Battelle 1974).

Retrievability/Recoverability. Waste disposed of using the meltdown emplacement concept would be retrievable for a short period, but movement down into the ice and successful

deployment of the concept design would quickly render the waste essentially irretrievable. Recovery is also considered nearly impossible. Retrievability for the other two emplacement concepts is indicated in the discussions above.

6.1.5.3 Status of Technical Development and R&D Needs

Present State of Development

Ice sheet disposal is in the conceptual stage of development and an extensive R&D program would be required to implement an operational disposal system (EPA 1979 and DOE 1979). Current technology appears adequate for initial waste canister emplacement using the concepts described. Necessary transportation and logistics support systems could be made available with additional R&D. The capability of ice sheets to contain radioactive waste for long periods of time is at present only speculative, because of limited knowledge of ice sheet stability and physical properties. Verification of theories that support ice sheet disposal would require many years of extensive new data collection and evaluation.

Technological Issues to be Resolved

Key technical issues that would have to be resolved for development of the ice sheet disposal concept include:

Choice of Waste Form

- Behavior of glass or other waste forms under polar conditions
- Ability of container to withstand mechanical forces.

Design of Shipping System for Polar Seas

- Extremes of weather and environmental conditions expected
- Debarkation port design
- Ship design
- Cask design
- Recovery system for cask lost at sea.

Design of Over-Ice Transport

- Crevasse detector
- Navigational aids
- Ability to traverse surface irregularities, snow dunes, and steep ice slopes
- Maintenance of road systems
- Recovery system for lost casks.

Design of Monitoring for Emplaced Waste

- Location, integrity, and movement of emplaced canisters

- Radioactivity of water at ice-rock interface
- Hydrologic connections to open oceans and effects on ice stability.

In addition, there are serious issues connected with the ability to adequately predict long-term ice sheet behavior, including rates of motion within the sheet, the physical state and rates of ice flow, movement of meltwater at the base of the sheet, and the long-term stability of the total sheet.

R&D Requirements to Make System Operational

R&D requirements to resolve these issues may be grouped in terms of those related to the handling, transportation, and emplacement of the waste, and those related to obtaining basic information on ice sheets. In the former group, R&D would be required in the areas of waste forms (content, shape, and materials), transportation (shielding, casks, ships, aircraft, over-ice vehicles), facilities (port, handling, inspection, repair), and supply logistics (fuel, equipment, personnel requirements). Research needs applying to ice sheets would include determination of ice sheet movement and stability through geological/geophysical exploration and ice movement measurements, studies of ice flow mechanics including effects of bottom water layers, studies of global and polar climatology, and acquisition and analysis of meteorological and environmental data.

Estimated Implementation Time and R&D Costs

If the ice sheet disposal concept were to prove viable, the time required to achieve an operating system is estimated to be about 30 years after the start of the necessary research program. The research program itself would require about 15 years of activity directed primarily toward improved understanding of ice sheet conditions, selecting an emplacement method, identifying and assessing ice sheet areas most suitable for the method selected, and research and preliminary development of systems unique to the particular emplacement method and site. Should the research program culminate in a decision to proceed with project development, an additional period of 12 to 13 years would be required to implement an operational disposal system.

R&D costs for ice sheet disposal are estimated to be \$340 million (in 1978 dollars) for the initial research and preliminary development program and between \$570 million and \$800 million for development, depending on the emplacement mode chosen.

Summary

Major uncertainties, shortcomings, and advantages of the concept are summarized below:

- The environment involved is non-benign to men and equipment, and the transportation limitations are severe.
- Understanding and performance assessment of the subsurface mechanisms of transport and package degradation are not developed to any degree.
- The concept does have the capacity for multiple barriers.

- The capability for corrective action over a long period is uncertain, and site selection criteria and performance assessment capability are nonexistent.
- No site is currently, or potentially in the future, available to the U.S. for R&D.

6.1.5.4 Impacts of Construction and Operation (Preplacement)

Health impacts, both radiological and nonradiological, and natural system impacts are analyzed below.

Health Impacts

Radiological impacts would in many ways be similar to those for mined geologic disposal but would have the added problem of extensive interim storage. Nonradiologic impacts might occur both as a result of routine operations or in abnormal or accidental conditions.

Radiological Impacts. Ice sheet disposal would be different from the mined geologic repository and other alternatives because of the requirement for extensive interim storage of either processed waste or spent fuel. Such storage would be necessary because lead times for research, development, and testing are 10 to 30 years longer than those for geologic disposal (DOE 1979). During this time, radiological effects would include doses to occupational personnel, the normal release of radioactive effluents to the atmosphere, and the potential for accidental release of radioactivity. At this time, no studies are available that provide a quantitative estimate of these impacts; however, it is expected that they would be similar to those from fuel storage facilities.

Preparation of waste for ice sheet disposal would be similar to that for mined geologic disposal methods. Likewise, the radiological effects associated with this option are assumed to be similar to those associated with geologic disposal methods. The radiological risks and impacts from the transportation of the waste would be to the Arctic or Antarctic essentially the same as those discussed in seabed disposal. The ice sheet disposal option is not sufficiently developed to estimate the radiological effects of routine operations on the ice sheet.

Accidents while unloading at the ice shelf seaport or during transport over the ice could create retrieval situations that would be difficult in the polar environment. Quantitative estimates of the radiological impact of such accidents are not available.

Nonradiological Impacts to Man and Environment. Potential nonradiological impacts could occur during all phases of ice sheet disposal operations. As with many of the alternative disposal strategies, impacts can be categorized as to whether they would occur during waste preparation, transportation, or emplacement activities. In general, those impacts associated with transportation and emplacement would warrant the most analysis. Waste preparation impacts would be similar to those for other disposal strategies discussed earlier.

Occupational casualties from the nonpolar activities are expected to occur at rates typical of the industrial activities that would be involved, and to be independent of both the nuclear and polar aspects of the remainder of the system. Operations are routinely carried out with nuclear systems and in the polar regions with safety comparable to that experienced in more familiar environments. In all likelihood, the required large-scale activities could also be performed safely, with the polar conditions being reflected in higher program costs rather than in decreased safety.

Accidents in processing and handling the waste material could occur before the material reaches the embarkation facility. Impacts resulting from such accidents are common to virtually all of the alternative disposal options. Other impacts would be virtually identical to those of the subseabed disposal option because in both cases the material would be transported to a coastal location.

Nonradiological health effects for activities that would occur on the ice sheet under abnormal conditions have not been studied extensively. Occupational impacts would occur, but as stated above, it is not expected that polar conditions will significantly alter the level of effects anticipated. Non-occupational effects would be even less significant, reflecting the lack of human activity on the ice sheets.

Natural System Impacts

Quantitative estimates of the radiological impact of ice sheet disposal on the ecosystem are not available. These impacts are expected to be small because there are very few living organisms in the polar regions, except along the coastline. Nonradiological ecological impacts at the disposal site are difficult to characterize because of a lack of understanding of the processes occurring in polar environments. The present understanding of impacts on the glacial ice mass or the dry barren valleys of Antarctica is limited. The effect of the heat that would be produced by the wastes on the ice or the potential geologic host media remains unclear.

Air impacts would result from the combustion products of over-ice transport vehicles, support aircraft, and fuel consumed for heating the facilities at the various sites. At present, the effects of these products are not considered a major problem.

Few, if any, ecological impacts are expected near the disposal sites because the plant and animal life are confined mostly to the coastal areas. Access routes and air traffic lanes could be made to avoid as much as possible the feeding, nesting, and mating spots of the birds and animals that inhabit the coastal areas. Fuel spills, equipment emissions, and general transportation support activities could lead to some localized impacts along the transportation disposal corridors. Few, if any, other impacts on water are expected, except for a marginal increase in temperature of the water that would be used for once-through cooling of canisters during sea transport. The only other water uses would be for consumption by the 200 operating personnel, which would be obtained by melting the ice.

Other possible land impacts considered in the reference study include accidental spills of fuel and the probability of fuel bladders rupturing during drop-offs. Rupture of the fuel bladders is considered to be a high risk because the fuel is capable of penetrating the snow and could reach the underlying ice where it would remain until evaporated or eventually buried by additional snow. Accidental spills could reach the ocean if the incident occurred near the edge of the ice sheet.

Socioeconomic Impacts

Socioeconomic impacts for the ice sheet disposal option would be similar to those for the island and subseabed disposal options. Because these options are still at the concept level, however, detailed socioeconomic assessments are not possible. In general, socioeconomic impacts would be experienced where handling facilities are constructed and operated.

Impacts that might be expected where handling facilities would be constructed include disruptions or dislocations of residences or businesses; physical or public-access impacts on historic, cultural, and natural features; impacts on public services such as education, utilities, road systems, recreation, and health and safety; increased tax revenues in jurisdictions where facilities would be located; increased local expenditures for services and materials; and social stresses.

The operating work force required for a dock facility would likely be comparable to that for any moderate-size manufacturing facility and impacts would vary with location. Impacts would be primarily in housing, education, and transportation, with no significant impacts on municipal services. Impact costs would presumably be offset by revenues, but socioeconomic considerations at this stage are not easily quantified.

Aesthetic Impacts

Aesthetic impacts are expected to be insignificant because of the remoteness of the area and the lack of permanent residence population (EPA 1979).

Aesthetic impacts for the ocean transportation activities and embarkation facilities would be very limited and similar to those of subseabed disposal. The waste packaging and transportation activities that would be a part of the ice sheet disposal process would have aesthetic impacts similar to those of mined geologic repositories. Noise, fugitive emissions, and the appearance of facilities and equipment used to prepare and transport the waste material are common to a number of disposal options. These impacts are generally reviewed in Chapter 4.

Resource Consumption

Predisposal activities would include packaging and transportation of spent fuel to seaports for shipment to the receiving port at the ice sheet, if spent fuel were disposed of rather than reprocessed waste. If reprocessing of spent fuel were undertaken, then predisposal activities would also include conversion of the waste to a high-integrity form, like

glass, before transportation to seaports. The resource requirements of these activities have been discussed elsewhere in this document for other disposal alternatives, and would be the same for ice sheet disposal, except for differences in transportation routings.

Little quantitative information exists on the energy, resource, and land requirements unique to ice sheet disposal. Ice sheet disposal would require construction of ships, airplanes, and over-the-ice vehicles that would not be required for other disposal alternatives. A greater number of shipping casks would also be required, because of the long cask turn-around time.

Transporting the waste material to its final destination across the ice fields would also require expenditure of energy. Either surface or air transport would use large quantities of fuel because of the great distances involved.

Some land impacts would probably be experienced in connection with the embarkation port facility. An area of about 1km^2 (0.4mi^2) would be required for the shielded cell and the loading dock facilities. The port facility would be equipped with its own separate water, power, and sewer systems to assure maximum safety. The over-ice transport routes would include an area at the edge of the ice sheet, ice shelf-edge, and ice-free areas on land for unloading the shipping casks. Approximately six support and fueling stations would be required along the transport route to the disposal area. Land requirements at the disposal site are estimated at $11,000\text{km}^2$ ($4,2000\text{mi}^2$) for waste from a plant producing 5 MTHM/day based on a waste canister spacing of one/Km.

International and Domestic Legal and Institutional Considerations

The ice sheet disposal option, like the island and seabed options, would require transporting waste material over the ocean, and the general international implications of such transportation are important.

Numerous legal and institutional considerations would emerge if the ice sheet disposal concept were seriously pursued in either Greenland or Antarctica. In the case of Greenland, treaty arrangements would have to be made with Denmark because Greenland is a Danish Territory.

In the case of Antarctica, a number of treaties and agreements exist that could affect the use of the ice sheets for storage and disposal of radioactive material. Disposal of waste in Antarctica is specifically prohibited by the Antarctic Treaty of 1959, of which the United States is a signatory (Battelle 1974). The treaty may be renewed after it has been in effect for 30 years, or amended at any time.

Outcomes of two meetings reflect the current range of international attitudes toward ice sheet waste disposal. One attitude was expressed in a resolution passed by the National Academy of Sciences, Committee on Polar Research, Panel on Glaciology, at a meeting in Seattle, Washington, May, 1973. The resolution neither favored nor opposed ice sheet waste disposal as such. However, a statement from a second meeting, on September 25, 1974, in Cambridge, England, attended by scientists from Argentina, Australia, Japan, Norway, the United Kingdom, the United States, and the USSR, recommended that the Antarctic ice sheet not be used for waste disposal.

6.1.5.5 Potential Impacts over the Long Term (Postemplacement)

Potential Events

Long-term impacts with the greatest potential significance are related to glacial phenomena that are not well understood. For example, ice dynamics and climatic variations affecting glaciation might be altered by waste disposal activities. Regardless of whether meltdown, anchored emplacement, or surface storage were used, potentially major modifications in the delicately balanced glacial environment could occur.

One of the major areas of uncertainty stems from our limited understanding of ice sheet conditions. Little is known of the motion of the continental ice sheets except for surface measurements made close to the coast (Gow et al. 1968). Three general types of flow have been defined--sheet flow, stream flow, and ice-shelf movement (Mellor 1959). Each type of flow appears to possess a characteristic velocity. It is also believed that ice sheets where bottom melting conditions exist may move almost as a rigid block, by sliding over the bedrock. Where there is no water at the ice-bedrock interface, it is believed that the ice sheet moves by shear displacement in a relative thin basal layer. The formation of large bodies of water from the waste heat could affect the equilibrium of such ice sheets.

In addition, two potential problems concerning the movement of the waste are unique to an ice sheet repository. First, the waste container would probably be crushed and breached once it reached the ice/ground interface as a result of ice/ground interaction. Second, the waste might be transported to the sea by ice movement.

Compared with other disposal schemes, the probability of human intrusion would be very low because the disposal area would be located in the most remote and inaccessible part of the world, presently with a low priority for exploration of natural resources or habitation. The lack of human activity in these areas would markedly decrease the chance of humans disturbing waste material emplaced in an ice sheet. Conversely, because of the remoteness of these areas they are relatively unexplored. Therefore they could attract considerable future resource exploration.

Potential Impacts

After the waste is emplaced and man's control is relinquished or lost, possible impacts fall into two broad categories. One of these relates to the reappearance of the radioactive waste in the environment, and the other involves the chance that the presence of waste would trigger changes in the ice sheets that would have worldwide consequences. For options that would place the waste within the ice or at the ice/ground interface, significant research would be required to predict future ice movements, accumulation or depletion rates, subsurface water flow rates, frictional effects at the interface, and trigger mechanisms. A major purpose of this research would be to compare the degree of sensitivity of the predicted behavior to man's ability to forecast long-term situations such as global weather patterns, stability of the ice sheets, and sea-level changes.

Specific areas of concern, as discussed below, are:

- Effects of waste on ice sheet environment
- Effects of ice sheet on waste
- Effects of waste on land environment.

Effects of Waste on Ice Sheet Environment. If waste canisters were allowed to reach or approach the bottom of the ice, they could possibly generate sufficient heat to produce a water layer over a large portion of the bottom surface of the ice. Furthermore, melt pools around the canisters could conceivably coalesce and also unite with any subglacial water, in the disposal area, to form a large water mass within the ice or at the edge of the ice-bedrock interface. Either event might trigger an increase in the velocity of the ice mass and perhaps produce surging. It has been postulated that major surges in the East Antarctica ice sheet could affect solar reflection and alter the sea level. The most extreme effect would be the start of glaciation in the Northern Hemisphere (Wilson 1964). The accelerated movement could also move emplaced material toward the edge of the ice sheet, possibly reducing the residence time. Basal ice sheet water could also conceivably form a pathway for transporting waste material from the disposal area to the edge of the ice sheet, and thus to the ocean.

Hypothetical dose calculations have been made for radionuclides released from an ice sheet disposal site into the ocean off the coast of Greenland (EPA 1979). On the basis of assumptions that a failure occurs in the disposal system, the release of radionuclides into the Greenland current of $8 \times 10^6 \text{ m}^3/\text{sec}$ would be 0.3 percent/yr of the total inventory available. Complete mixing could occur in the ocean. Human pathways are assumed to be mostly via fish consumption. The maximum dose was considered to be from an individual consuming 100 kg/yr of fish caught in these contaminated waters and is estimated to be 0.2 mrem/yr. Further discussion of radioactive releases to the ocean is included in Section 6.1.4.5 on the subseabed concept.

Effects of Ice Sheet on Waste. Movement of the ice sheet might cause shearing or crushing of canisters, allowing water to come in contact with the waste form so that leaching could occur. Such breakage would most likely occur when the canisters are moved along the ice-bedrock interface.

If major climatic changes were to produce an increase in temperature in the polar region, the ice sheet might erode to such an extent that it would allow the waste to be much closer to the edge of the ice. The temperature increase could also increase the velocity of the ice movement toward the coast.

Effects of Waste on Land Environment. As in the case of space and subseabed disposal, geologic repository facilities are assumed to be constructed for TRU and other wastes not disposed of through the procedures established for the majority of HLW. Long-term effects could result from these auxiliary activities. These impacts would be similar to those

described for the mined geologic concept. The other land area that could be impacted is the region of dry barren valleys in Antarctica. If wastes were placed in this area, impacts would be very similar to those of the mined geologic repository. The major difference would be that the ground-water regime in Antarctica would mostly affect remote frozen ground-water systems.

Terrestrial ecosystems in the ice sheet regions under study for disposal sites are limited in diversity. Severe climatic conditions limit most organisms to the seaward margins of both Greenland and Antarctica. Consequently, the potential for impact to terrestrial organisms in the ice sheet disposal is quite limited. Potentially more significant are the long-term ecological effects of any accidents that would occur on the land mass where the wastes were generated. As described in Section 5.6, these impacts should not be significant unless an accident or encroachment occurs.

6.1.5.6 Cost Analysis

The cost of depositing nuclear wastes in ice sheets is currently expected to be relatively high; higher, for example, than the cost of geologic emplacement in the U.S. This is primarily because of the high costs for R&D as presented in Section 6.1.5.3. Capital, operating, and decommissioning cost estimates are presented below.

Projected Capital Costs

Projected capital costs for ice sheet emplacement of 3000 MT/yr of spent fuel, or the wastes recovered from processing that amount of fuel, are \$1.4 billion to \$2.3 billion as shown in Table 6.1.17.

Projected Operating Costs

Projected operating costs for the emplacement of 3000 MT/yr of spent fuel or HLW are shown in Table 6.1.18.

Decommissioning Costs

Decommissioning costs associated with contaminated equipment would probably be limited primarily to the shipping casks used to transport waste canisters for ice sheet disposal. These costs are estimated at \$9.7 million, which is 10 percent of the initial capital cost of the shipping casks. Costs for decommissioning other facilities and equipment are assumed to be similar to those for other waste disposal alternatives.

6.1.5.7 Safeguard Requirements

Because the reference concept uses both ice sheet and mined geologic disposal, its implementation would require safeguarding two separate disposal paths. The risk of diversion for the meltdown concept would be basically a short-term concern because once the waste had been successfully disposed of in accordance with design, it would be considered irretrievable. For the anchored and surface storage concepts, although the waste would be considered retrievable for as long as 400 years, the harsh environment in which it would be

TABLE 6.1.17. Capital Costs For Ice Sheet Disposal
(Millions of 1978 Dollars)

Case I. Meltdown or Anchored Emplacement: Surface Transportation

1. Construction of Port Facilities	730
2. Sea Transport Vessels	290
3. Ice Breakers	190
4. Over-Ice Transport Vehicles	100
5. Drilling Rigs	50
6. Monitoring Equipment	50
7. Shipping Casks	100
8. Aircraft	100
9. Support Facilities	<u>150</u>
	1760

Case II. Surface Storage

1. Construction of Port Facilities	730
2. Sea Transport Vessels	290
3. Ice Breakers	190
4. Over-Ice Transport Vehicles	100
5. Surface Storage Facility	500
6. Monitoring Equipment	50
7. Shipping Cask	100
8. Aircraft	100
9. Support Facilities	<u>190</u>
	2250

Case III. Meltdown or Anchored Emplacement: Aerial Emplacement

1. Construction of Port Facilities	500
2. Sea Transport Vessels	150
3. Aircraft	500
4. Shipping Casks	100
5. Monitoring Equipment	50
6. Support Facilities	<u>150</u>
	1450

TABLE 6.1.18. Operating Costs For Ice Sheet Disposal
(Millions of 1978 Dollars/Year)

Emplacement Concept Emplacement Method	<u>Meltdown or Anchored</u>		<u>Surface Storage</u>
	<u>Surface</u>	<u>Aerial</u>	<u>Surface</u>
Cost Category:			
Operating Personnel(a)	34	29	39
Material & Consumables(b)	58	29	58
Services & Overhead(c)	68	58	78
Capital Recovery(d)	<u>175</u>	<u>141</u>	<u>224</u>
Total	335	257	399

(a) Based on \$50,000/man-year.

(b) Including \$29 million/yr and \$5 million/yr port upkeep for surface and aerial emplacement, respectively.

(c) Based on twice the operating personnel costs.

(d) Based on 10 percent of capital expenditures (not including research and development costs). Encapsulation costs not included.

placed and the equipment needed for retrieval would also make any risk of diversion primarily a short-term concern. Only minimum safeguards would be required after emplacement. Physical protection of the sensitive facilities and transportation operations would be the most effective way to deny access to the waste for the short term, as is common to most waste disposal alternatives. See Section 4.10 for additional discussion of predisposal operation safeguard requirements.

6.1.6 Well Injection

6.1.6.1 Concept Summary

Well injection technology was initially developed by the oil industry for the disposal of oil field brines. These brines were usually pumped back into the original reservoir and, in some cases, used to "drive" the oil toward a producing well. The well injection concept has subsequently been used for the disposal of various natural and industrial wastes. The techniques developed in the oil industry handle liquid wastes only - particulate matter can cause blocking of the pores in rock.

A well injection process using grout was developed by Oak Ridge National Laboratory (ORNL) for the injection of remotely handled TRU liquid radioactive wastes into shale strata (ERDA 1977). This technique is also suitable for grout slurry wastes, and a new facility is now under construction at ORNL for liquid and slurry waste injection (ERDA 1977). Well injection could be a low cost alternative to deploy and operate because of the widespread use of the required techniques and the "off-the-shelf" availability of the main components. Two reference methods of well injection are considered in this section: deep well liquid injection and shale grout injection.

Deep well injection would involve pumping acidic liquid waste to depths of 1,000 to 5,000 m (3,300 to 16,000 ft) into porous or fractured strata suitably isolated from the biosphere by overlying strata that are relatively impermeable. The waste may remain in liquid form and might progressively disperse and diffuse throughout the host rock. This mobility within the porous host media formation might be of concern regarding release to the biosphere. Questions have also arisen regarding the possibility of subsequent reconcentration of certain radioisotopes because of their mobility. This could lead to the remote possibility of criticality if, for instance, the plutonium is reconcentrated sufficiently. Isolation from the biosphere would be achieved by negligible ground-water movement in the disposal formation, particularly towards the surface, retention of nuclides due to sorption onto the host rock mineral skeleton, and low probability of breaching by natural or man-made events. The concept is not amenable to a multiplicity of engineered barriers.

For shale grout injection, the shale would first be fractured by high-pressure water injection and then the waste, mixed with cement and clays, would be injected into suitable shale formations at depths of 300 to 500 m (1,000 to 1,600 ft) and allowed to solidify in place in layers of thin solid disks. The shale has very low permeability and probably good sorption properties. The injection formations selected would be those in which it could be shown that fractures would be created parallel to the bedding planes and would therefore remain within the host shale bed. This requirement is expected to limit the injection depths to the range stated above. Direct operating experience is available at ORNL for disposal of TRU wastes by shale grout injection. The grout mixes have been designed to be leach resistant and hence the concept minimizes the mobility of the incorporated radioactive wastes.

Isolation from the biosphere is achieved by low leach rates of radionuclides from the hardened grout sheet, negligible ground-water flow particularly up through the shale strata, retardation of nuclide movement by minerals within the shale strata, and low probability of breaching by natural or man-made events.

6.1.6.2 System and Facility Description

System Options

The two reference concepts for well injection disposal of nuclear waste have been selected from a number of options available at each step from the reactor to disposal at the well injection facility. These two concepts are judged as "most likely" based on the status of current technology. A summary of various options to be considered for well injection disposal is illustrated in Figure 6.1.18. Additional pertinent data available on the options can be found in various source material listed in Appendix M.

Waste-Type Compatibility

For both reference concepts the waste form injected would be HLW. Since disassembly and some processing would be necessary for well injection, the concepts would be suitable for fuel cycles that recycle uranium and plutonium. However, well injection could also be applied to once-through fuel cycles after dissolution or slurring of spent fuels. In these

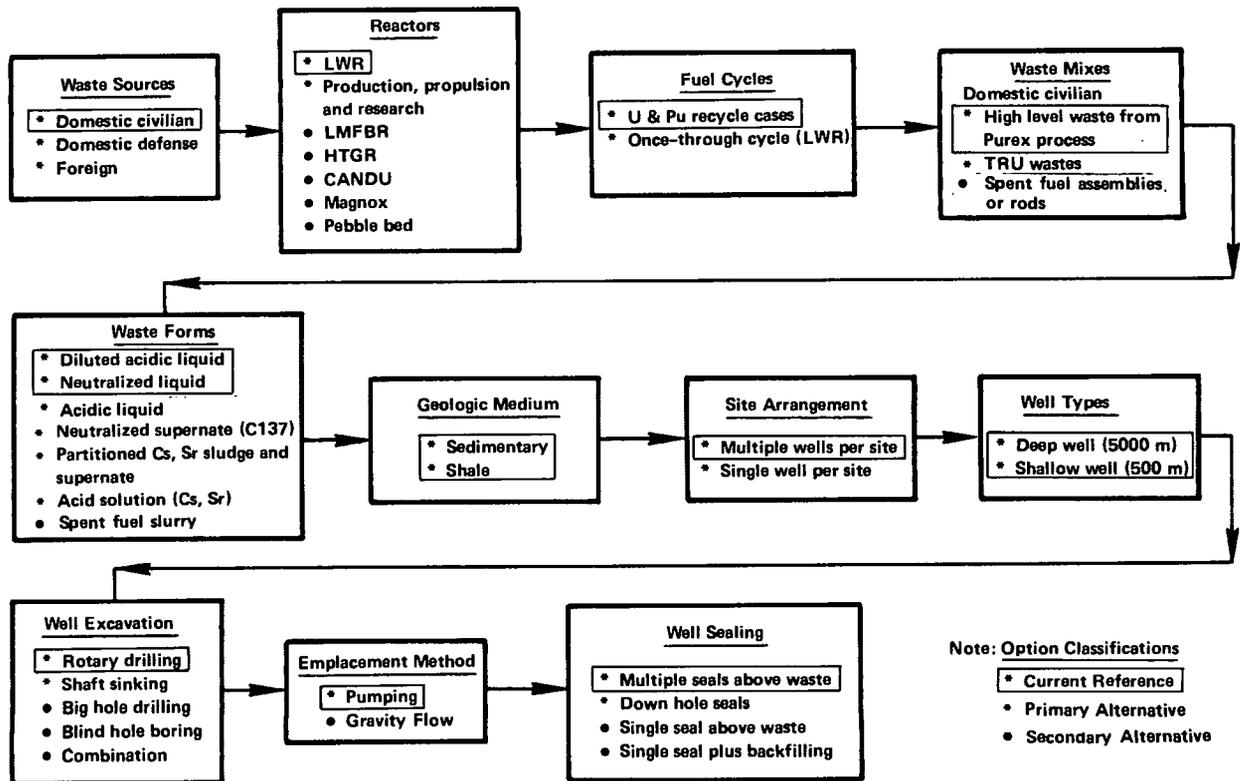


FIGURE 6.1.18. Major Options for Well Injection Disposal of Nuclear Waste

cases, the injection liquid would contain large amounts of actinides, which might affect the thermal properties and interaction mechanisms of the waste in the host media. Well injection might also be used to dispose of high-heat-level partitioned wastes, which could relieve high thermal loadings in a mined geologic repository for example. Note that retrieval would be difficult and incomplete using either concept, although deep well injection would have more potential for at least partial retrieval than would the shale-grout method, which would fix the waste in a relatively insoluble solid.

For deep well injection, the liquid waste would have to be substantially free from all solid matter to prevent clogging of the formation pores. Filtration down to 0.5 μ m particles is typical for process waste injection systems (Hartman 1968). The waste would have to remain acidic to ensure that all the waste products stay in solution.

For shale grout injection, neutralized waste (sludge and supernate) would be mixed with cement, clay, and other additives.

Waste System Description

The fuel cycle and process flows associated with the two reference options are illustrated on Figure 6.1.19. Significant features of these concepts are summarized in Table 6.1.19.

Both concepts are based on restricting the maximum temperature in the injection formation to 100 C (212 F), assuming a geothermal gradient of 15 C/km (44 F/mile), to avoid undesirable mineralogical effects that would occur at higher temperatures. (For example, comparatively large amounts of waste would be released from the clay mineral montmorillonite if

TABLE 6.1.19. Reference Concepts Summary (DOE 1979)

Reference Concepts	Depth of Injection	Disposal Formation
Deep well liquid injection	100-m-thick zone at average depth of 1,000 m	Sandstone with shale caprock at 950-m depth; porosity 10 percent
Shale grout injection	100-m-thick zone at average depth of 500 m	Shale extending to within 50 m of ground surface

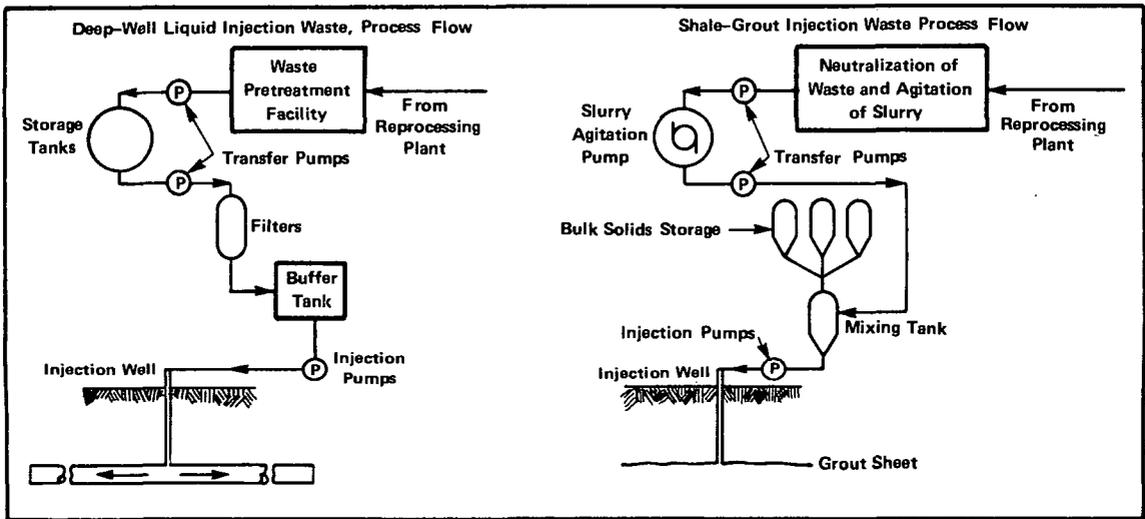
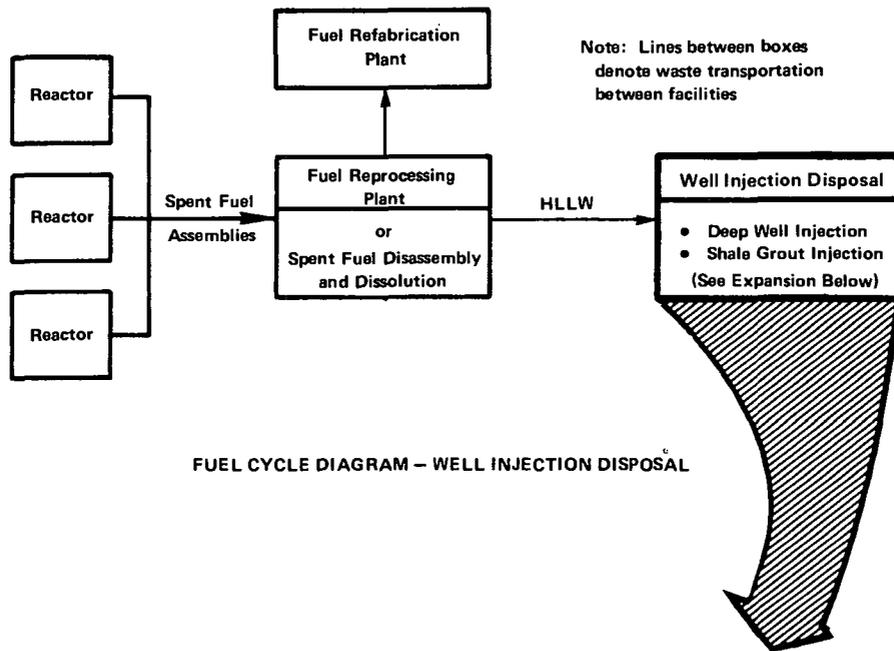


FIGURE 6.1.19. Waste Management System--Well Injection Disposal

heated to above 100 C) (EPA 1973). Although disposal strata containing more inert minerals, particularly quartz-rich sandstones suitable for deep well liquid injection, might sustain higher temperatures, thermal effects on containment formations, which may include temperature sensitive minerals, would also have to be considered.

Deep Well Injection

In the deep well injection concept, the liquid wastes would be fed into porous or fractured strata, such as depleted hydrocarbon reservoirs, natural porous strata, or zones of natural or induced fractures. To protect freshwater aquifers from waste contamination, the injection zone would have to be well below the aquifers and isolated by relatively impermeable strata, e.g., shales or salt deposits.

In general, injection requires pressure at the wellhead, although in some circumstances gravity feed is sufficient. The controlling factors are the rate of injection and the permeability of the disposal formation. The increase in the total fluid volume in an injection zone is accommodated by compression of any fluid already present and expansion of the rock formation. The relation between injection rates and pressures is based on extensive oil-well and ground-water experience. Injection is possible at depths down to several thousand meters.

For this concept, the activity of the injection waste has been assumed to be controlled by the allowable gross thermal loading, the injection zone thickness, and the porosity in that zone. It is also assumed that one injection zone with two wells would be used at each site. In the long term, the waste might progressively disperse and diffuse throughout the host rock and eventually encompass a large volume. The concentration might be variable and unpredictable. Thus, criteria for permissible activity levels might be required. Determination of the dilution requirement is complicated by the sorption of nuclides onto the mineral skeleton, to an extent determined by waste chemistry and rock mineral content. If sorption were too high, concentration of heat-generating components might result in "hot spots".

Injected waste might be partially retrieved by drilling and pumping, but sorption of nuclides onto the mineral skeleton and precipitation within the pores would limit the amounts recovered.

Predisposal Treatment. In deep well injection, spent fuel would be shipped to a processing facility at the well injection site. The spent fuel would be dissolved in acid and the hulls removed. (For recycle, the uranium and plutonium would be removed from the acid solution.) The acid solution would constitute the basic waste form for isolation.

The acid waste from reprocessing would contain both fission products and actinides. Between 60 and 75 percent of the heat generated in the initial emplacement years would be due to ^{90}Sr and ^{137}Cs . Partitioning strontium and cesium from the remainder of the waste

would permit different isolation practices to be adopted for the high-heat-generating, relatively short-lived isotopes (half-lives about 30 years) and the remainder of the waste containing the much longer lived, lower heat generating isotopes.

The liquid waste would be diluted with water or chemically neutralized and pumped from the reprocessing facility to the injection facility or to interim storage in holding tanks.

Site. Deep well injection would require natural, intergranular fracture porosity or solution porosity formations, overlain by impermeable cap rock, such as shale. A minimum acceptable depth for disposal would be about 1,000 m (3,300 ft) (EPA 1973). The injection site must not conflict with either present or future resource development.

Synclinal basins would be particularly favorable sites for deep well liquid injection since they consist of relatively thick sequences of sedimentary rocks frequently containing saline ground water (Warner 1968). Ground-water movement within the injection formation would have to be limited, however, particularly vertical movement.

The lithological and geochemical properties of the isolation formation would have to be stable so that the behavior of the waste could be accurately predicted. In general, sandstone would be the most suitable rock type because it combines an acceptable porosity and permeability with chemically inert characteristics relative to the acid waste form.

The overall site area has not been determined yet, but would be greater than the 1270 ha (3140 acres) initial injection area and would depend on the maximum horizontal dimension of the injection area, the size of control zone required around the repository, and the total amount and type of waste to be injected.

Drilling System. The drilling rigs would be similar to those used in the gas and petroleum industries and would be portable for movement from one location to another on the site. Each complete rig would require a clear, relatively flat area, approximately 120 m x 120 m (400 ft x 400 ft) at each hole location (see Section 6.1.1).

Repository Facilities. The processing plant would be located on site as an integral part of the overall injection system. The basic repository facilities would be similar to those required for the very deep hole concept, as discussed in Section 6.1.1 (Bechtel 1979a).

Interim storage tanks similar to those described for the rock melt concept (Section 6.1.2) would be provided for surge capacity. The stainless steel tanks would have a combined capacity of about 10^6 liters (2.8×10^5 gal) which equals 3 months production. The tanks would be similar in design to those at the AGNS plant in Barnwell, South Carolina, which are contained in underground concrete vaults and provided with internal cooling coils and heat exchangers to prevent the waste from boiling.

An underground pipeway system would connect the reprocessing facility to the storage tanks and the injection facility. The pipe would be double cased and protected by a concrete shielding tunnel with leak detectors provided in the annulus of the pipe. The pipeway design would provide containment, monitoring, decontamination, maintenance, and decommissioning

capabilities, primarily performed remotely. A heavy concrete and steel confinement building would provide containment for the well and injection operations and shielding for the radioactive systems.

Sealing Systems. The well hole would probably be sealed by a combination of borehole seals and backfilling, using a procedure similar to the one discussed for the very deep hole concept (Section 6.1.1).

Retrievability/Recovery. Liquid waste that had been injected might be partially retrievable by conventional well techniques. Although much of the waste might be physically or chemically sorbed by host geologic media, some species, in particular, ^{137}Cs , would be expected to remain in at least partially retrievable solution.

Shale Grout Injection

In the shale grout injection process, neutralized liquid waste or an irradiated fuel slurry would be mixed with a solids blend of cement, clay, and other additives, and the resulting grout would be injected into impermeable shale formations. The initial fracture in the shale would be generated by hydrofracturing with a small volume of water. The injection of waste grout into this initial fracture would generate sufficient pressure to propagate a thin horizontal crack in the shale. As injection of the grout continued, the crack would extend further to form a thin, approximately horizontal, grout sheet, several hundred feet across. A few hours after injection, the grout would set, thereby fixing the radioactive wastes in the shale formation. Subsequent injection would form sheets parallel to and a few feet above the first sheet.

The principal requirement for shale grout injection is that the hydrofracture, and hence the grout sheet, develops and propagates horizontally. Vertical or inclined hydrofractures could result in the waste gaining access to geologic strata near the surface, and even breaking out of grout at the bedrock surface itself. Theoretical analyses indicate that, in a homogeneous isotropic medium, the plane of hydrofracture develops perpendicularly to the minor principal stress (NAS 1966). Thus, a requirement for horizontal hydrofracturing is that the horizontal stresses exceed the vertical stresses.

On the basis of work at ORNL, approximately 40 injection wells would be required at each of five facilities. The activity level for the shale grout injection alternative is based on the reference concept (Schneider and Platt 1974) of 40 Ci/l activity in the initial grout. The acceptable gross thermal loading (GTL) could be assured by controlling the number of grout injections in the disposal formation. Depending on the fuel cycle, the maximum number of 2-mm (0.08-in.)-thick grout layers would be five to seven per injection site.

Site. A thick sequence of essentially flat-lying shale strata would be required for shale grout disposal, with in situ stress conditions favorable for the propagation of horizontal hydrofractures. Such conditions are generally found to a maximum depth of 500 to 1,000 m (1,650 to 3,300 ft). As with deep well liquid injection, the site would have to be located to preclude conflicts with resource development.

Shale deposits in the United States have been studied for suitability for underground waste emplacement (Merewether et al. 1973). The studies conclude that shale, mudstone, and claystone of marine origin in areas of little structural deformation, low seismic risk, and limited drilling are generally most promising. These include the Ohio shale of Devonian age in northern Ohio and the Devonian-Mississippian Ellsworth shale and the Mississippian-coldwater shale in Michigan. In the Rocky Mountain states, the Pierre shale and other thick shales of late Cretaceous age are also potential host rocks.

The overall site area for shale grout injection has not been determined yet, but it would be greater than the 1270 ha (3140 acres) initial injection area and would depend on the maximum horizontal dimension of the injection area and the size of the control zone required around the repository.

Drilling System. The drilling system for shale grout injection would be similar to that for deep well injection.

Repository Facilities. Repository facilities for shale grout injection would be identical to those for deep well injection with the exception of additional high-pressure pumps for fracturing and equipment related to mixing the grout with the liquid waste prior to injection (see Figure 6.1.19).

Sealing Systems. The repositories would be sealed in the same manner as deep well holes.

Retrievability/Recovery. Wastes disposed of by this concept would be essentially irretrievable because of the fast solidification and stability of the waste-grout mixture. Total recovery of the wastes would likely involve extremely difficult and extensive mining operations to excavate the rocklike waste form.

6.1.6.3 Status of Technical Development and R&D Needs

Present State of Development and Technological Issues

The basic techniques required for well injection of fluids and grouts have been developed in the course of many projects undertaken by the oil and chemical industries for the disposal of nonradioactive toxic and nontoxic wastes. In addition, limited disposal of radioactive waste grouts has been successfully completed at ORNL (ERDA 1977, Delaguna et al. 1968).

Geology. The geology of sedimentary basins in the United States has been examined extensively with a view to suitability for deep well liquid injection of radioactive wastes, and reports are available covering several areas.^(a) In addition to these studies, a large

(a) See Repenning 1962, Sandberg 1962, Beikman 1962, MacLachlan 1964, Legrand 1962, Repenning 1959, Colton 1961, and DeWitt 1961.

volume of geologic data (stratigraphy, lithology, petrography) exists for potential disposal areas. These data have been gathered for basic geologic research or as a result of resource exploration and exploitation. However, the existing data are considered suitable for only conceptual, generic studies and identification of candidate sites.

Geohydrology. Modeling to predict waste extent and nuclide transport would be required for both liquid and grout injection. In the past decade, numerical modeling methods using finite-difference and finite-element techniques have been developed using available high-speed digital computers (Pinder and Gray 1977, Remson et al. 1971). Two- and three-dimensional fluid-flow techniques with thermal and stress dependency are available. Computer codes also exist for the analysis of radionuclide transport, including the effects of decay, adsorption, and dispersion (Burkholder 1976). However, these analytical techniques are limited because of an insufficient data base and incompletely defined constitutive parametric relationships.

State-of-the-art testing techniques include the use of multiple devices to isolate sections of the borehole. These devices provide for reduction in measurement error through improved control of bypass leakage. The multiple devices also help determine directional permeability (Maini et al. 1972). Multiple hole analyses are used to define the direction and magnitude and measure of rock mass permeability (Rocha and Franciss 1977, Lindstrom and Stille 1978). Because rock properties are directionally dependent, particular consideration must be given to methods of analyzing field data before a well injection site could be chosen.

Drilling and Injection Technology. The well injection disposal would require relatively simple engineering design, construction, and operation. Oil well drilling technology, fundamental to the concept, is available and well proven.

The deep well injection disposal method has been applied in the United States for natural wastes, in particular, oil-field brines, and for industrial wastes, such as steel pickle liquors, uranium mill wastes, and refinery and chemical process wastes^(a). The deepest waste injection well completed and operated to date was at Rocky Mountain Arsenal, where fractured Precambrian gneiss, at a depth of 3,660 m (12,000 ft), was used as the disposal formation (Pickett 1968).

Shale grout injections of remotely handled TRU wastes have been carried out at ORNL at a depth of about 275 m (900 ft) (ERDA 1977). Over 6.8×10^6 l (1.8×10^6 gal) of waste containing primarily ^{137}Cs (523,377 Ci) with a lesser amount of ^{90}Sr (36,766 Ci), together with minor quantities of other radionuclides have been injected over 10 years.

(a) Such applications are described in DeWitt 1961, Pinder and Gray 1977, Remson et al. 1971, Burkholder 1976, Maini et al. 1972, Rocha and Franciss 1977, Trevorrow et al. 1977, Lindstrom and Stille 1978, White 1965, Hult et al. n. d., Pickett 1968, Warner and Orcutt 1973, Lunn and Arlin 1962, Clebsch and Baltz 1967, Spitsyn et al. 1973, Capitant et al. 1967, and Roedder 1959.

Waste Preparation Technology. Liquid waste might require pretreatment to ensure compatibility with the rock. No operating injection facilities exist at present for high-level acid wastes. Pretreatment for most industrial wastes comprises filtration and limited chemical treatment. Since well injection is usually being pursued to reduce waste processing requirements, chemical treatment is minimal, and may include the addition of biocides and chloride to prevent plugging of the well from bacterial growth (Hartman 1968).

Waste preparation for shale-grout injection at ORNL has been the subject of extensive testing to develop an economical mix with good pumping and leach-rate characteristics (Moore et al. 1975, Hollister and Weimer 1968). Research indicates that the use of ash as a partial substitute for cement reduces costs and enhances strontium retention. Mixes incorporating various clays and grout shale have been tested. Leach rates of 3.2×10^{-5} g/cm²/day for strontium and 2.1×10^{-6} g/cm²/day for cesium have been obtained. The latter value is approximately equivalent to the leach rate for borosilicate glass (ERDA 1977).

Isolation and Safety. Isolation and safety analyses are based on

- Definition of source term (concentration, form, location, time)
- Characterization of pathway (transport velocity, chemical or physical changes, path length barriers, ecosystems involved)
- Exposure and "dose-to-man" calculations for both specific groups and total population.

A range of data values for the parameters can be analyzed to provide a probabilistic basis for the results. Methods involving modeling and analysis of failure processes have been employed for analyzing the performance of conventional disposal options (Logan and Berbano 1977) and would also be applicable to deep well injection concepts.

R&D Requirements

Since experience in the basic techniques required for well injection exists, the uncertainties associated with the design basis are related primarily to extrapolation of this experience to other waste forms, to other geologic settings, and to modified quantities and disposal rates. There are already techniques for preparing radioactive wastes in liquid or slurry form; however, there are uncertainties in formulating liquid wastes that would provide stability and compatibility with the disposal formation. For slurries, further R&D would be required for the development of optimum mixes, which would be related to the specific characteristics of the waste and disposal formation.

Geologic formations suitable for the injection of waste would have to be identified and verified on a site-specific basis. The exploratory techniques needed to do this are in an early stage of development, and would require further R&D with particular emphasis on verifying local geologic structure, establishing local and regional geohydrologic conditions, determining thermal and mechanical properties and in situ stresses, and locating and orienting discontinuities.

With the basic technology for injecting radioactive wastes into geologic strata already available, these research and development requirements can be categorized into several discrete areas of development, as described below.

System Data Base. It would be essential that the total R&D program be supported by a data base that covered all the components that could affect performance of the disposal system. The data base would cover the waste form, its modification, storage and injection, and the characteristics of the disposal formation from near to far field.

Development of Criteria for and Categorization of Siting Opportunities. The two types of well injection disposal methods, liquid and grout injection, would require significantly different but clearly definable disposal formation characteristics. Disposal site selection would have to proceed in stages, starting with the derivation and assembly of specific criteria, followed by successive narrowing of the field of choice to a specific site or sites. This approach would provide valuable generic hydrogeological data at an early stage for subsequent use in other R&D studies. The selection process could be undertaken initially using available geologic and hydrologic data and techniques. At the site-specific level, however, the use of yet-to-be developed "nonpenetrative" techniques might be required to minimize the amount of down-hole exploration.

Liquid and Slurry Wastes. A key facet of well injection is pretreatment of the liquid or slurry to a form that would be both compatible with the receiving formation and also the best use of the potential of that formation to fix and retain the nuclides. Optimum forms and requisite admixtures would have to be identified. The R&D program would have to proceed from the generic to the specific when the geochemistry of the disposal formation is known.

Techniques for Predicting the Configuration of Injected Wastes. Fundamental to the concept of "safe" disposal of waste is the necessity to predict, with a high degree of accuracy, the configuration that the injected wastes, whether liquid or grout-fixed slurry, would adopt in the disposal formation for both the short and long term. The technology should provide this capability.

For the liquid injection method, predictive capability is currently limited by the existing data base. Numerical simulation techniques are available, but these do not cover the range of conditions that might be encountered. Mathematical models for geohydrological and geochemical interaction studies would be needed.

"Nonpenetrative" Exploration Techniques. The presence of a drill hole could impair the isolation of a disposal site. At present, the majority of exploratory techniques require drilling at least one hole (and often several) to obtain reliable information from geological strata. R&D would be needed to develop nonpenetrative exploration techniques, similar to other geologic disposal methods.

Sealing Systems. It is assumed that the sealing system for well injection would have to meet the same time requirements for sealing penetrations that a mined repository must meet. The primary purpose of the seal is to inhibit water transport of radionuclides from the waste

to shallow ground water or to the surface for an extended time period. Expansive concretes make the best seals under current technology and do so at an acceptable cost. However, current experience with seals, whether of cement, chemical, or of other materials, is only a few years old. Further development of sealing technology would, therefore, be required (Bechtel 1979a). For integrity to be maintained, the sealing material would have to meet the following requirements:

- Chemical composition - the material must not deteriorate with time or temperature when compared to host rock characterization.
- Strength and stress-strain properties - the seal must be compatible with the surrounding material, either rock or casing.
- Volumetric behavior - volume changes with changes in temperature must be compatible with those of enclosing medium.

The sealing system for well injection would consist not only of plugs within the casing, but also of material to bridge the gap between casing and competent rock not damaged by drilling. To minimize possible breaks in containment, rigorous quality assurance would be required during emplacement of several high quality seals at strategic locations within the borehole.

Research and development would be needed in two major areas - material development and emplacement methodology - to ensure complete isolation. Material development would include investigating plugging materials (including special cements), compatible casing materials, and drilling fluids. Because the seal would include the host rock, these investigations should include matching of plugging materials with the possible rock types. It is conceivable that different materials would be required at different levels in the same hole.

Emplacement methodology would have to be developed for the environment of the hole. Considerations would include operation in the aqueous environment, casing and/or drilling, and fluid removal. Because the emplacement methodology would depend on the type of material, initial studies of material development would have to precede emplacement methodology development. However, the two investigations would be closely related and would interface closely. In situ tests would have to be performed to evaluate plugging materials. Equipment developed would include quality control and quality assurance instrumentation.

Monitoring Techniques. In common with other methods of underground disposal, techniques would be required for monitoring the movement/migration of radioactive material from the point of emplacement.

Borehole Plugging Techniques. Borehole plugging techniques would require development at an early stage to permit safe exploration of candidate sites.

Implementation Time and Estimated R&D Costs

The R&D program described above is generic. Specific estimates for required implementation time and R&D costs would depend on the details of the actual development plan, and are deferred pending plan definition.

Summary

Major uncertainties, shortcomings, and advantages of the concept are summarized below:

- The concept is not compatible with the multi-barrier philosophy, relying only on a potentially non-inert waste form and the geology.
- Performance assessment and siting technology for HLW injection are essentially non-existent.
- Retrievability, technical conservatism, and adequate design margins do not appear possible due to the diffuse nature of the emplaced material.
- The emplacement technology is considered to be essentially available.

6.1.6.4 Impacts of Construction and Operation (Preemplacement)

In some respects the environmental impacts of the well injection concepts are better understood than the impacts from the other disposal alternatives. This is because of their current use--deep well by the oil and gas industry to dispose of chemical waste and shale grout injection by the Oak Ridge National Laboratory to dispose of remotely handled TRU wastes. Potential use of well injection for disposing of long-lived or high-level radio-active waste, however, has not been demonstrated.

Although quantitative estimates of environmental impacts of well injection have not been made, it is expected that many of the impacts would be essentially the same for the two reference concepts.

Health Impacts

Radiological Impacts. The radiological impacts from routine operations during most phases of well injection disposal (e.g., reactor spent fuel storage, and intermediate spent fuel storage) are expected to be the same as those for a mined geologic repository. However, the extra operation to reprocess spent fuel from the once-through fuel cycle to produce a liquid solution or grout could be expected to add to the radiological impacts. Quantitative estimates of these impacts are not available at this time. Likewise, the radiological impacts associated with the transportation of wastes are expected to be similar to those for a mined geologic repository, with the exception of transporting HLW from the reprocessing plant. Since, for the reference repositories, the injection facility is adjacent to the reprocessing plant, the need to transport HLW is eliminated, which thereby reduces the corresponding radiological impact.

Unavoidable environmental effects of the well injection option would include operational radiation doses to facility workers involved in injection or maintenance and repair. Design and operational procedures would be directed to reducing doses to the lowest levels possible. At the ORNL remotely handled TRU waste facility the radiation exposure per man per grout injection has averaged 0.025 rem during injection operations and 0.188 rem during preinjection maintenance (ERDA 1977). However, the data are not sufficient to determine whether these occupational exposures would be applicable to an HLW repository. Accident scenarios

may be conveniently divided into surface and subsurface events. Surface operating accidents would include pipe ruptures and spills, failure of transfer or injection pumps, and loss of necessary cooling to the storage tanks. To minimize risk, normal nuclear engineering design strategies would be required, with redundancies incorporated into all critical systems and components (for example, pumps, power supply, and monitoring equipment). Subsurface accidents, for which contingency plans would have to be prepared, would include well-pipe rupture, equipment failures, uncontrolled fracture development (shale grout injection), and penetration of waste through the containment formation due to highly permeable features, abandoned or poorly sealed wells, or exploration or monitoring of drill holes. Site exploration and analyses would be directed toward minimizing the probability and the effects of subsurface failures.

Presently, there are no quantitative estimates of the radiological impacts of such accidents to occupational personnel, nonoccupational personnel, or the ecosystem. Furthermore, since the waste would be in a nonsolid form for well injection, the radiological impacts are not expected to be similar to those resulting from accidents at a mined geologic repository.

Nonradiological Impacts. Little formal study has been completed on the nonradiological health effects of the well injection disposal process. In general, predisposal activities, such as fuel handling, storage, transportation, and reprocessing, for both reference concepts would be the same as for a mined geologic repository. Pretreatment of the disposal formation with acid, however, might be required. Although potential impacts have not been quantitatively assessed, it can be concluded that nonradiological health effects would result from handling this hazardous material.

Because wastes injected into the wells would have to be in liquid or grout form, two important differences are anticipated between well injection and mined geologic disposal. First, the well injection disposal site would have to be at the same place as the reprocessing facility. Colocating these facilities would minimize the transportation requirements and associated risks. It would also reduce some of the nonradiological impacts associated with transportation activities.

Second, well injection would involve surface and subterranean activities with different hazards than those associated with mined geologic disposal--formation drilling and fracturing, compared to large-scale excavation, are the principal below-ground activities that could lead to nonradiological health impacts. Preparing the wastes for disposal would involve facilities designed to mix the wastes with clay, cement, and other additives for the shale-grout method. For the liquid injection process, more limited mixing facilities would be needed. In either case, studies completed to date have not identified significant nonradiological impacts for these activities under routine operating conditions. Under abnormal conditions, pipe ruptures and spills, failure of injection pumps, and other problems discussed under radiological impacts could lead to nonradiological impacts as well.

Natural System Impacts

Effects on the ecosystem near a well injection disposal site would be similar to those associated with any heavy engineering project. In considering these impacts, it must be remembered, however, that the disposal site would include reprocessing and disposal facilities.

Ecological impacts from these processes are categorized into preconstruction and post-construction activities. Initial construction activities would involve clearing vegetation, drilling, and geophysical surveying. Impacts of these initial activities would affect vegetation, soil, water, and other resources to varying degrees depending on the characteristics of the specific site being developed. Impacts of this type of activity are evaluated for specific sites.

Construction impacts would include those of a reprocessing facility, as described in Chapter 4. Construction of facilities to prepare the wastes for injection, as described above, would also be needed.

Postconstruction, or operational, nonradiological ecological impacts would be more limited than those of preconstruction and construction activities. Many operational activities would occur below the surface. Ecological impacts from these activities could occur if some of the fluids injected into the well were to enter the ground-water system and were transported to the biosphere or otherwise affected aquatic resources. Surface runoff or material spilled on the surface could also cause localized ecological impacts.

Socioeconomic Impacts

Socioeconomic effects from constructing and operating a well injection repository would be felt most intensely in the immediate vicinity of the facility. In general, impacts would be representative of those of a major engineering facility. No quantitative data exist on the construction or operational employment requirements of a well injection disposal system. Impacts, however, should be similar to those described for the very deep hole concept (see Section 6.1.1.6). In addition, socioeconomic impacts associated with the reprocessing facility would be felt at the disposal site. These impacts are discussed in Section 4.7. In analyzing these discussions, it must be remembered that colocation would lead to a greater concentration of impacts at the disposal site, but at the same time would reduce the number of separate nuclear facilities constructed.

Aesthetic Impacts

Aesthetic impacts for the well injection disposal option would be similar to those of other subsurface disposal methods except for the presence of the reprocessing facility at the disposal site. Again, collocating facilities could increase the impacts at the chosen site, but the fact that only one site is needed suggests an overall reduction in aesthetic impacts.

Aesthetic impacts could be accurately assessed only within the context of a specific site. In a general context, however, aesthetic impacts related to drilling and other geologic activities are covered in the aesthetic impact discussions for mined geologic

repositories (Section 5.5) and the very deep hole concept (Section 6.1.1.6). Aesthetic impacts of reprocessing facilities are discussed in Section 4.7.

Resource Consumption

Suitable well injection sites would be sedimentary basins, which are frequently prime areas for fossil fuels. However, after the wastes had been safely emplaced, geologic exploratory activities in the vicinity of the site would have to be restricted. It has been suggested that potentially usable minerals from the zone of influence of the repository would be inventoried before implementation would begin. On the other hand, the disposal zone itself could be considered a resource for which alternative uses might be found, for example, storage of freshwater or natural gas.

Other resources consumed in the well injection process would include energy for transportation, processing, and disposal. Land would be required for the reprocessing and disposal facilities. For the shale-grout disposal method, clay, cement, and other materials would be needed. No critical material, other than fuel, would be consumed by well injection disposal.

International and Domestic Legal and Institutional Considerations

Implementation of the well injection option would require two important policy decisions that could be shaped by institutional forces. First, the process does not lend itself to handling spent fuel from reactors. Processing would be needed to transform this material into a form that could be readily injected into the well. The reprocessing approach most often proposed contravenes the current U.S. position against reprocessing. This would have to be resolved before well injection disposal could be implemented.

The second policy decision stems from the need to locate the disposal facility and the fuel reprocessing plant at the same site. Although such a system would be effective in limiting liquid waste transportation, it is likely that neither facility would be optimally located. It would have to be decided whether the benefits of well injection disposal outweigh potential disadvantages of such collocation. Obviously, such a decision would have to be made in light of domestic institutional considerations.

Another aspect of the well injection concept that could foster concern is the need to obtain records of previous drilling activities. States typically maintain such records and generally oversee drilling programs. If this disposal option were implemented, information would be needed and procedures would have to be established to evaluate data from adjacent well sites. The relationship between existing regulatory activities and the well injection disposal process would have to be defined prior to implementation.

Aside from the issues outlined above, the legal and institutional considerations of this option would be similar to those of the mined geologic repository discussed in Section 5.5.

6.1.6.5 Potential Impacts Over Long Term (Postemplacement)

An unavoidable long-term impact of well injection waste disposal is that alternative storage or disposal applications for the site are eliminated. Examples of possible uses are

natural gas storage, freshwater storage, and disposal of other wastes of lower or shorter-lived toxicity. In addition, as noted earlier, exploration for natural resources and subsequent mining in a large area around the disposal facility would be subject to control. The extent of exclusion and limited activity buffer zones would depend on the characteristics of the disposal formation, and in particular, its hydrologic and geochemical conditions. Finally, evidence exists that injection of wastes into certain formations could potentially lead to seismic activity and earthquakes.

Potential Events

Natural Events. The long-term leaching and transportation of radionuclides in the ground-water system to the biosphere would be a fundamental pathway in the well injection concept, as it is with all geologic concepts. Assessment of the environmental impact would require predictive modeling of the rock mechanics, hydrology, and geochemistry of the disposal and containment formations, together with an adequate data base to characterize the biosphere. The disposal area would be selected to minimize the risks from seismic and volcanic activities and their effect on the hydrologic regime. Seismic events could induce tectonic effects within the disposal area, causing permeability and flow changes. Volcanic activity could result in catastrophic breach of the containment formation, or could generate unacceptable, thermally induced flow patterns. The risk of meteorite impact would be similar to that for a mined geologic repository; however, with deep-well liquid disposal, the waste would be in a more mobile form. The impact of gross changes, such as climate variations or polar ice melting, would, in general, depend on their effect on the hydrologic regime. Increased erosion (because of glaciation, for example) could reduce the cover of the disposal formation.

An impact of potentially major significance is the increased chance of an earthquake that could result from injecting waste material into rock formations. A relationship between deep well liquid injection and increased seismicity has been suggested (Evans 1966) in connection with earthquakes at Denver and injection at the Rocky Mountain Arsenal well. Other studies (Hollister and Weimer 1968, Dieterich et al. 1972) have shown that deep well injections in the Rocky Mountain Arsenal Range have been instrumental in producing seismic events. Obviously, such concerns are significant and would have to be seriously evaluated for specific sites. Knowledge of the in situ stress state for both concepts would be needed before proceeding with the well injection option because of the chance of earthquakes developing. The depth of shale grout injection would be limited by the requirement that vertical stresses be less than horizontal stresses.

Manmade Events. Exclusion and controlled-use buffer zones would be set up around an injection facility. Nevertheless, the risks associated with drilling into a waste-liquid or grout disposal formation would have to be considered. Changes in the surface and subsurface hydrologic regime of the area, because of reservoir construction, deep excavation and construction, and resource exploitation outside the buffer zone, would require analysis.

The geologic formation in which a well injection repository would be located would have to be bounded by impermeable strata and free of water-transmitting faults. Such formations occur in the sedimentary basins in the U.S., and it is these basins that oil and gas companies are exploring for petroleum and natural gas. This exploration could cause a major safety problem by connecting waste disposal zones with aquifers.

Potential Impacts

As with the mined geologic repository, the principal pathway for release of radionuclides to the biosphere in the long term would be by ground-water transport. It is believed, however, that the likelihood of ground water reaching the injected waste is extremely small.

The only quantitative estimates on the movement of radionuclides via ground water transport are from ORNL's experience with grout injection of remotely handled TRU waste into shale (ERDA 1977).

The maximum quantity of activity that could be leached from a single grout sheet was calculated, using data presently available (ERDA 1977). This sheet would have a volume of about 28,300 m³ (1 million ft³) and could contain as much as 500,000 Ci of ⁹⁰Sr (if a maximum waste concentration of 5 Ci/gal is assumed) and an equal amount of ¹³⁷Cs. Leach data indicate that the 6-month leach rate of radionuclides from cured grouts would not exceed 6.2 x 10⁻⁵ Ci/month of ¹³⁷Cs per sq ft of leached area, 1.7 x 10⁻³ Ci/month-ft² of ⁹⁰Sr, 5.5 x 10⁻⁷ Ci/month-ft² of ²⁴⁴Cm, and 5.6 x 10⁻¹⁰ Ci/month-ft² of ²³⁹Pu. If the entire grout sheet surface were exposed to water flow, a maximum of 62 Ci/month of ¹³⁷Cs, 1700 Ci/month of ⁹⁰Sr, 0.6 Ci/month of ²⁴⁴Cm, and 6 x 10⁻⁴ Ci/month of ²³⁹Pu would be leached. If the water flow is assumed to be 0.5 ft/day, the calculated concentration of ²³⁹Pu in the water would be approximately 1 x 10⁻⁶ Ci/ml (less than the concentration guide for this isotope in uncontrolled areas). The shale surrounding the grout sheets has considerable ion-exchange capacity for cesium and strontium; a calculation yields rate of movement of leached cesium and strontium through the shale that would be so low that these nuclides would be transmuted by radioactive decay long before they approached the surface. The small quantity of ²⁴⁴Cm that might be leached would also be retained by the shale.

6.1.6.6 Cost Analysis

Capital, operating, and decommissioning costs of well injection disposal have not been estimated. However, since well injection disposal would not require costly mining operations, it could offer a low-cost means of disposal compared to mined repositories.

Cost data are available from ORNL (ERDA 1977) for a site-specific application of grout injection disposal of RH-TRU. Estimated capital costs for a new waste shale fracturing disposal facility, adjusted to 1978 dollars, are \$6.0 million. Annual operating costs are estimated at \$110,000. No data are given for decommissioning costs. The costs are estimated

for a facility to perform removal of large volumes of mobile radioactive wastes from existing near-surface storage facilities at Oak Ridge.

6.1.6.7 Safeguard Requirements

Because of the restrictions concerning the transportation of high-level liquid waste, which require the injection facility to be collocated with the fuel reprocessing plant, the accessibility to sensitive materials would be extremely limited. However, this waste disposal system would probably be used in a uranium-plutonium recycle fuel cycle so there would be incremental increases in accessibility in other parts of the fuel cycle similar to most recycle scenarios. In addition, the difficulty of retrieving material once it had been successfully disposed of would increase the difficulty of diversion and the waste form (liquid) would complicate the transportation and handling problems for a potential diverter. The deep well injection repository would require additional safeguards since at least partial retrieval by drilling and pumping might be possible. Material accountability would be enhanced by ease of sampling and measurement of liquids, but gross accountability (i.e., gallons vs canisters) would be slightly more difficult than for the reference mined geologic concept.

See Section 4.10 for additional discussion of predisposal operations safeguard requirements.

6.1.7 Transmutation

6.1.7.1 Concept Summary

The primary goal of waste disposal has been stated as protection of the public. This would be achieved in mined geologic disposal by containing the high-level radioactive waste for the time period during which it retains significant quantities of potentially harmful radionuclides. One alternative to this approach is to selectively eliminate the long-lived radionuclides by converting or transmuting them to stable or short-lived isotopes. This approach would shorten the required containment period for the remaining waste. Shortening the containment period would increase confidence in predicting the behavior of the geologic media and reduce the requirements on the isolation mechanism. Thus, an attractive feature of transmutation is that it has the potential to reduce the long-term risk to the public posed by long-lived radionuclides.

In the reference transmutation concept, spent fuel is reprocessed to recover the uranium and plutonium. The remaining high-level waste stream is partitioned into an actinide stream and a fission product stream. The fission product stream is concentrated, solidified, vitrified, and sent to a terrestrial repository for disposal. In addition, actinides are partitioned from the TRU-contaminated process waste streams from both the fuel reprocessing plant and the mixed oxides fuel fabrication plant. The waste actinide stream is combined with recycled uranium and plutonium, fabricated into fuel rods, and reinserted into the reactor. For each full power reactor year, about 5 to 7 percent of the recycled waste actinides are transmuted (fissioned) to stable or short-lived isotopes. These short-lived isotopes are separated out during the next recycle step for disposal in the repository. Numerous recycles result in nearly complete transmutation of the waste actinides.

A disposal system that uses transmutation would have the environmental and health impacts associated with the recycle of uranium and plutonium and with the partitioning of the actinides from the waste stream. If uranium and plutonium recycle were adopted for other reasons transmutation would be more feasible but would still involve additional impacts. For example, highly radioactive fuel elements containing recycled waste actinides would need to be fabricated, handled, and transported. The additional facilities and waste treatment processing steps required could be expected to increase effluent releases to the environment, the occupational exposure, the risk of accidents, and costs. Since only about 5 to 7 percent of the recycled waste actinides would be transmuted to stable isotopes in each reactor irradiation, numerous recycles would be required with attendant additional waste streams.

6.1.7.2 System and Facility Description

System Options

The reference concept was selected from several available options. These options are listed in Figure 6.1.20 for each major step in a flowsheet using transmutation.

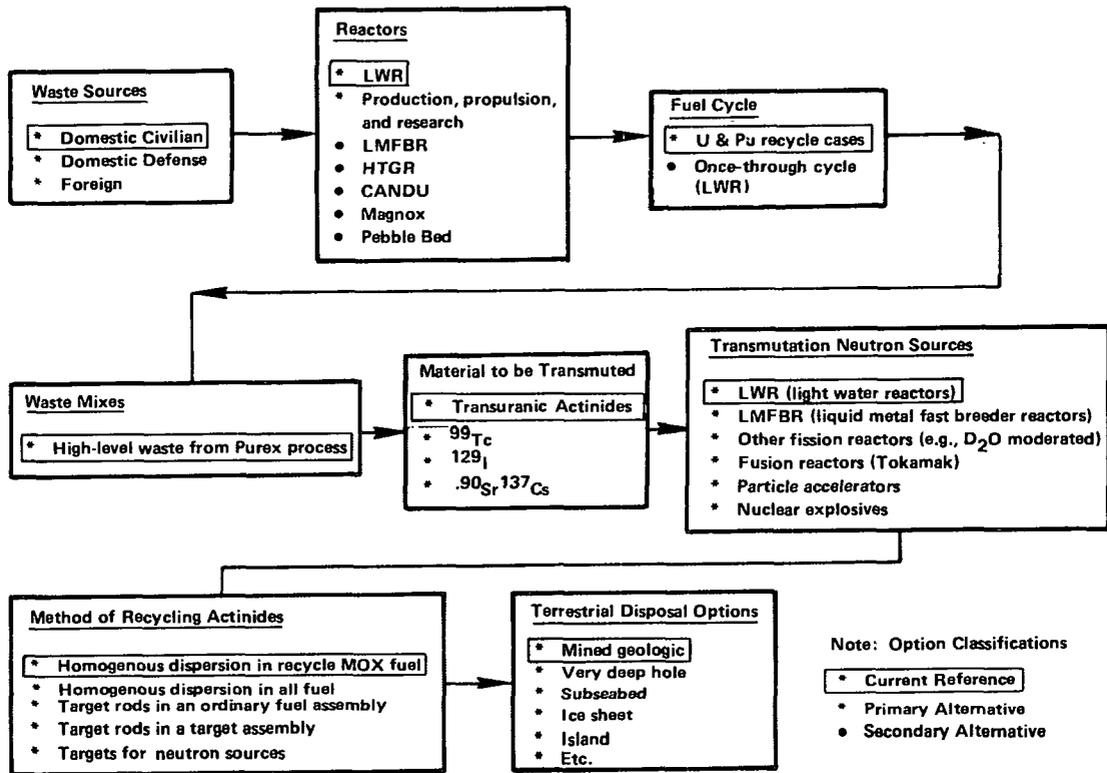


FIGURE 6.1.20. Major Options for a Waste Disposal Alternative Using Transmutation

The reference concept was selected somewhat arbitrarily to be used as a basis for comparison and to help identify the impacts associated with a typical transmutation fuel cycle. If transmutation were selected as a candidate alternative for further research and development, considerable study would be required to optimize the available alternatives. Additional information concerning the advantages and disadvantages of the many process options is available in sources listed in Appendix M.

Waste-Type Compatibility

Transmutation would be applicable to only those fuel cycles that involve the processing of irradiated nuclear fuel, e.g., the recycle of uranium and plutonium. In that context, transmutation would not apply to once-through fuel cycles. It could be used with both commercial and defense waste, although little work has been done concerning defense wastes.

Waste-System Description

The fuel cycle and process flow for the reference concept are shown in Figure 6.1.21. The cycle begins with the insertion of a reload of fuel into the reactor. The reload is two-thirds fresh enriched UO_2 and one-third recycle mixed oxide (MOX) fuel, which has all the waste actinides (i.e., neptunium and other transplutonics) homogeneously dispersed in it.

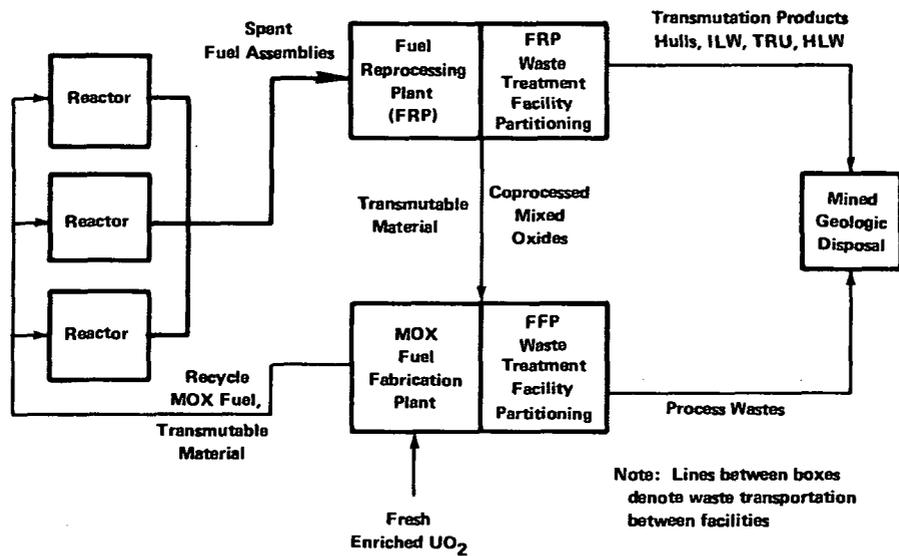


FIGURE 6.1.21. Partitioning-Transmutation Fuel Cycle Diagram

The cycle continues by:

- Irradiating the reload to a burnup of 33,000 Mwd/MTHM
- Discharging and decaying the reload for 1-1/2 years
- Reprocessing the UO₂ and MOX fuels together
- Sending the TRU-contaminated wastes to the fuel reprocessing plant waste treatment facility (FRP-WTF) for partitioning
- Returning the recovered TRU and the TRU-depleted wastes to the reprocessing plant
- Combining the recovered actinides with the processed MOX and transporting the mixture to the refabrication plant, after a 6-month delay
- Adding sufficient uranium to the MOX product to achieve the desired end-of-cycle reactivity. (This product is in powder form and contains the waste actinides.)
- Refabricating the MOX product
- Sending the TRU-contaminated wastes from refabrication to the fuel fabrication plant waste treatment facility (FFP-WTF) for partitioning
- Returning the stream of recovered actinides to the fabrication plant
- Incorporating the recovered actinides with MOX recycle streams within the facility
- Sending TRU-depleted wastes to a mined geologic repository.

Simultaneously, the fresh enriched UO₂ fuel is fabricated in a separate facility. At this point, the cycle is completed with the fabricated fuels being inserted into the reactor. The details of the waste treatment facility (WTF) process and plant design are given in Tedder et al. (1980) and Smith and Davis (1980).

Predisposal Treatment

In a fuel cycle involving transmutation, it would be necessary to partition the materials to be recycled and transmuted. The partitioning flowsheet would have two fundamental steps. The first would be to separate the actinides from other materials and the second would be to recover the actinides in a relatively pure form. Actinides would be separated by various methods and would originate from many sources, including high-level waste, dissolver solids, cladding, filters, incinerator ashes, salt wastes, and solvent cleanup wastes. The extractable actinides from these operations would be sent to actinide recovery, where they would be partitioned and purified.

Facilities Description

There are four facilities in the reference fuel cycle that process the actinides: the fuel reprocessing plant (FRP), the fuel fabrication plant (FFP), and a colocated waste treatment facility (WTF) for each. The purpose of the two WTF's would be to recover a high percentage of the actinides that would ordinarily be delegated to process wastes.

The FRP-WTF and FFP-WTF would have the following common process capabilities:

- (1) Actinide recovery
- (2) Cation exchange chromatography (CEC)
- (3) Acid and water recycle
- (4) Salt waste treatment
- (5) Solid alpha waste treatment.

In addition, the FRP-WTF would have high-level liquid waste and dissolver solid waste treatment process capabilities. The WTF facilities would be constructed on sites about 460 m (1,500 ft) from the FRP and FFP, but still within a fuel cycle center that would allow common services and utilities for the entire center. Additional detailed design and cost information is available in Smith and Davis (1980).

Since transmutation would take place in the reactor itself, no special facilities would be required, although the irradiation levels of the recycle fuel require that the fuel assemblies be handled remotely. Because transmutation would eliminate only a specific segment of the waste, all the facilities required for conventional terrestrial disposal, e.g., a mine geologic repository as described in Chapter 5, would also be necessary in this fuel cycle. The use of transmutation would not significantly change the total amount of waste or the necessary throughput of waste disposal facilities.

Retrievability/Recovery

The segment of waste disposed of in the mined geologic repository would exhibit the same characteristics discussed in Chapter 5 of this report.

6.1.7.3 Status of Technical Development and R&D Needs

Only the referenced use of transmutation - recycling, using commercial nuclear reactor fuels, to minimize the actinides contained in radioactive waste - is discussed here. Part of the R&D associated with transmutation would be the continued investigation of other useful applications of the process. There are several other waste constituents that could be transmuted.

Present Status of Development

Transmutation represents an advanced processing concept that would require R&D work before incorporation into any system. There are still uncertainties associated with many of the subsystem details. Although the concept is technically feasible, it should be recognized that the required design bases have not been sufficiently refined to permit construction of full-scale facilities. For some partition subsystems, laboratory experiments have been developed to demonstrate technical feasibility only. Only preliminary material balance calculations have been performed and, in most cases, no energy balances are available.

A number of transmutation devices for converting various nuclides to other more desirable forms have been studied. Neutron irradiation can be carried out with nuclear explosive devices, fission reactors, or fusion reactors. Accelerators can provide charged particle beams of protons or heavier ions for producing neutrons for irradiating selected nuclides. For the actinides, the most practical transmutation occurs by irradiation by a fission reactor neutron source. The estimated actinide transmutation rate utilizing commercial light water reactors is about 6 percent for each full-power year that the actinides are in the reactor (EPA/MITRE 1979).

There are four principal methods for recycling actinides in light water reactors: (1) dispersing the actinides homogeneously throughout the entire fuel reload, (2) dispersing the actinides homogeneously in only the mixed-oxide fuel, (3) concentrating the recycled waste actinides in target rods within an otherwise ordinary fuel assembly, and (4) concentrating the recycled waste actinides in target rods that are then used to make up a target assembly. In the first two methods, the actinides include all of the plutonium generated in the reactor. In the second two methods, plutonium (an actinide) is excluded from the targets but is recycled in a mixed-oxide fuel. On the basis of preliminary qualitative evaluation, it would appear that the second recycle mode, homogeneous dispersal of the actinides in the mixed-oxide fuel, is preferred over the others (Wachter and Croff 1980).

Technological Issues

The effect of a transmutation recycle, as opposed to the uranium and plutonium recycle mode, on the various elements of a conventional fuel cycle depends largely on two factors--the transmutation rate in the reactors and the manner in which the transmutation reactors are decommissioned as the cycle is eventually terminated. Important technological issues are:

- The use of commercial power reactors as transmutation devices might result in fissile penalties, reactor peaking problems, reduced reactor availability, and increased operating costs.
- Because of increased concentrations of radioisotopes with high specific activities, and/or modifications of existing systems due to changes in requirements, transmutation recycles could require additional containment systems to limit the release of radioactivity at the reactor site to acceptable levels.
- Many transmutation cycles would increase fuel handling requirements because of the more frequent insertion and removal of fuel and transmutation targets from the reactor core. Most transmutation cycles would result in increased shielding requirements both for fresh and spent fuels and transmutation targets.
- Decommissioning and disposal of reagents from partitioning and transmutation facilities would be complicated by the increased demands for shielding, multiple chemical processes, and waste streams.

The duration of the transmutation cycle is important in estimating its overall effectiveness in reducing the total radiotoxicity of transmutable elements in the environment. Premature termination of the transmutation cycle could actually increase the radiotoxicity of the wastes. This is because the resulting inventory sent to a final disposal system might have more activity than it would if the transmutation cycle had not been initiated.

R&D Requirements

The R&D requirements for partitioning would involve specific near-term subtasks to clarify points of uncertainty in the current process parameters and techniques. However, to fully develop and demonstrate actinide partitioning, a program would have to include additional process research and development, a cold (nonradioactive) testing facility, equipment development and testing, and pilot plant design, licensing, construction, testing, and operation.

Transmutation R&D would include specific nuclide cross section measurements, reactor physics calculations, and irradiation to full burnup of test fuel assemblies to verify calculations. The irradiation tests would also serve to confirm the design and fabrication of the fuel assemblies and their compatibility with and performance in the reactor during power operation.

The design, construction, and testing of a prototype shipping cask made from the relatively unconventional materials proposed might also be required. Specific aspects of cask technology that might require attention are: techniques for industrial fabrication of special shielding materials, such as B_4C/Cu and LiH , investigation of the ability of the cask using such materials to conduct the heat from the fuel contents, and the effect of the unusual construction materials on safety considerations in cask design.

Finally, continuing overall studies to define the preferred methods of operating the fuel cycle and the impacts and benefits of this operation would be of primary importance.

Implementation Time

The long lead time for implementing this alternative is based on the orderly development of a commercial scale partitioning plant, which would be expected to take about 20 years. The first 10 years would be devoted to partitioning research and the development and testing of a pilot plant, as reflected in Table 6.1.20. All of the R&D programs involving transmutation, fuel assembly and shipping cask development, and system studies could be accomplished in concurrence with the partitioning schedule.

Estimated R&D Costs

Table 6.1.20 identifies estimated R&D costs necessary to demonstrate the transmutation of actinides. It does not include costs associated with providing a commercial scale partitioning plant, the necessary modifications to the fuel fabrication facility and light water reactors, or a transportation system required to utilize the partitioning-transmutation of actinides as a waste disposal alternative.

Summary

Major uncertainties, shortcomings, and advantages of the concept are summarized below:

- The concept is actually a method of waste treatment or conversion to a more benign form; it is not an independent disposal method.
- Additional waste streams during the process are generated so that the actual volume of waste for isolation is greater than without it.
- The technology for efficient transmutation (waste partitioning and advanced reactors) are considered to be long-term achievements.

TABLE 6.1.20. Estimated Transmutation R&D Costs And Implementation Time

	<u>Cost, \$ million</u>	<u>Time Span, years</u>
Partition R&D (Includes Pilot Plant)	560	10
Transmutation R&D	16	15
Fuel R&D	80	15
Transportation	56	10
System Studies	8	Continuous

6.1.7.4 Impacts of Construction and Operation (Preemplacement)

As described in Section 6.1.7.1, the transmutation option would include elimination of certain long-lived radioactive wastes and the disposal of the remaining waste material in a mined geologic repository. The potential benefits of transmutation that would be realized for the lower levels of long-lived hazardous material are discussed in Section 6.1.7.5, while short-term impacts of construction and operation are discussed here. Because these short-term impacts include those of a mined geologic repository, impacts identified in Section 5.6 must be considered a part of this option. In addition, impacts associated with reprocessing and discussed in Section 4.7 would occur.

Because transmutation is a waste processing option involving extra waste treatment steps, a meaningful impact analysis is possible only when a transmutation system is compared with a reference processing and disposal system. In the following analysis, the reference system includes waste reprocessing and final disposal in a mined geologic repository.

Another important factor in this discussion is that impacts attributed to one plant generally relate to a reprocessing plant handling 2000 MTHM per year and a fuel fabrication plant handling 660 MTHM per year. Such a hypothetical plant provides the basis of much of the information used in this analysis (Blomeke et al. 1980, Fullwood and Jackson 1980, Logan et al. 1980). Depending on the actual amount of nuclear wastes generated, several of these plants could be constructed.

Health Impacts

Radiological Impacts. The increased frequency of waste handling and transportation activities associated with the transmutation option suggests that it would result in increased radiation exposures compared with the mined geologic repository option.

ORNL estimated the radiological occupational impact of the reference concept based on routine exposure, maintenance exposure, and anticipated abnormal occurrences (Fullwood and Jackson 1980). Table 6.1.21 presents the collective dose rates calculated for the four facilities included in the study. The values range from a low of 3 man-rem/plant-year for an abnormal occurrence in the FFP-WTF to a high of 230 man-rem/plant-year for routine and maintenance exposure in the FFP.

The radiological exposure to the general public arising from routine operations is a consequence of the fact that the facilities would have to provide fresh air for the workers and vent gases to the atmosphere. In spite of elaborate air-cleaning practices and equipment, small amounts of radioactive materials would be discharged into the atmosphere; the amount varying with the chemical species. Estimates have been made for the amounts of radioactive materials that are expected to be discharged from each plant (Fullwood and Jackson 1980). The resulting exposures, based on these estimates, are presented in Table 6.1.22. The values range from 680 to 736 man-rem/plant-year for the Reference Facility and the P-T respectively.

TABLE 6.1.21. Annual Routine Radiological Occupational Dose

Facility	Exposure, man-rem/plant-year		
	Routine	Operation Maintenance	Abnormal
FRP (1)	220	220	10
FRF-WTF (2)	220	220	10
FFP (3)	230	230	10
FFP-WTF (4)	90	90	3
Reference Facility (1) and (3)			
P-T	(1-4)		

The more significant of the postulated accidents have been analyzed as to the resulting effects on the plant workers. In general, individual worker exposure would exceed public exposure because of closeness to the accident. Isotopic differences between the two cycles would result in small differences in exposure, so there is negligible distinction between the Reference and the P-T cycle, except that the Reference Facility does not contain the two WTF's. The totals for the component facilities are presented in Table 6.1.23. The details of the accidents and other assumptions are given in Fullwood and Jackson (1980).

Table 6.1.24 presents corresponding data for the non-occupational consequences of the postulated accidents.

TABLE 6.1.22. Annual Routine Non-Occupational Dose

Process Stage	Exposure, man-rem/plant year	
	Ref. Facility	P-T
FRP	680	730
FRP-WTF	-	5.3
FFP	7×10^{-3}	1.7×10^{-2}
FFP-WTF	-	0.55
Totals	680	736

TABLE 6.1.23. Occupational Radiological Exposure--Abnormal

<u>Facility</u>	<u>Conditions</u>	<u>Exposure,</u> <u>man-rem/plant year</u>
	FRP	
FRP-WTF		1.3×10^{-2}
FFP		4×10^{-2}
FFP-WTF		7×10^{-3}

Besides the plants and processes another major activity in the fuel cycle would be transportation links for fresh fuel movement, spent fuel movement, powder movement between the FRP and FFP, and waste movement from the FRP-FFP complex to the repository and disposal area. Table 6.1.25 presents data resulting from accident analyses of the six transportation steps considered for the two fuel cycles.

Nonradiological Impacts. Nonradiological impacts would result from two factors that are unique to the transmutation alternative. First, the partitioning process would require additional facilities at the reprocessing plant and at the MOX fuel fabrication facility. Second, the nature of the wastes that would be generated by transmutation dictates increased transportation activities.

TABLE 6.1.24. Non-Occupational Radiological Exposures--Abnormal

<u>Process Stage</u>	<u>Exposure, man-rem/plant year</u>	
	<u>Ref. Facility</u>	<u>P-T</u>
FRP	5×10^{-3}	5×10^{-3}
FRP-WTF	-	6×10^{-5}
FFP	3×10^{-5}	3×10^{-5}
FFP-WTF	-	6×10^{-5}
Reference Facility	5×10^{-3}	
P-T		5.2×10^{-3}

TABLE 6.1.25. Transportation Non-Occupational Radiological Exposures--Abnormal

Transportation Step	Exposure, man-rem/plant year	
	Ref. Facility	P-T
Spent Fuel	2.3×10^{-3}	3×10^{-3}
Powder	2.3×10^{-10}	3×10^{-10}
Fresh Fuel	6×10^{-5}	3×10^{-5}
Cladding Hulls	1.2×10^{-2}	1.3×10^{-2}
HLW	8×10^{-4}	6×10^{-4}
NM-HLW	1×10^{-1}	9.8×10^{-2}
Totals	1.1×10^{-1}	1.1×10^{-1}

A closer examination of the first factor reveals that the additional partitioning facilities would be colocated at reprocessing and fuel fabrication sites. These incremental changes are analyzed as they would affect operational, environmental, and resource considerations.

Regarding the second factor, transportation impacts, the relatively small carrying capacity of the canisters that would be used to transport the fresh and spent fuel means more trips per unit of fuel than with options involving unpartitioned wastes. Furthermore, more waste would be generated. This would lead to more transportation impacts. It is estimated that the facilities included in this option would process 2,000 MTHM per plant per year. This means an estimated nine trips involving hazardous material would have to be made each day, as compared with an estimated seven trips per day for fuel reprocessing without transmutation (Fullwood and Jackson 1980). Although the increased emissions, chance of derailment, and community concern associated with more intensive transportation could not be accurately determined until a specific disposal system is proposed, it is recognized that transportation impacts would be greater than those for the reprocessing-only case.

Nonradiological health effects would occur as a result of construction and operation activities. In spite of scrubbers and other air-cleaning devices, small amounts of hazardous materials would be discharged into the atmosphere. There would be two main sources of these pollutants: the chemical processes themselves and the auxiliary services, primarily the steam supply system, which is assumed to burn fuel oil. Table 6.1.26 presents the annual health effects for transmutation. The data are based on estimates for the Allied General Nuclear services plant at Barnwell, South Carolina, but are scaled to allow for the larger size of the transmutation facilities. The health effects were estimated from epidemiological studies on SO₂ and its relationship to the other pollutants.

The increased transportation required for the transmutation alternative suggests a greater likelihood of occupational and nonoccupational hazards than with options not involving partitioning. Unlike radiological impacts, nonradiological concerns should not vary significantly from those of an industrial facility not involved in nuclear activity.

TABLE 6.1.26. Summary Effects (Per Plant-Year) of Non-Radiological Effluents (Fullwood and Jackson 1980)

Plant	Premature Deaths/yr		Permanent Disabilities/yr ^(a)	
	Reference Facility	Transmutation	Reference Facility	Transmutation
FRP	4	4	14	14
FRP-WTF	--	7	--	21
FFP	0.2	0.2	0.6	0.6
FFP-WTF	--	3	--	9
Totals	<u>4.2</u>	<u>14.2</u>	<u>14.6</u>	<u>44.6</u>

(a) Based on disabilities lasting longer than 6000 person-days.

Probably the single most important nonradiological hazard would result from the chemical processing, handling, and transportation activities, during which accidents could happen. The uncertainties associated with this unproven technology make precise analyses of these hazards difficult. Health evaluations, however, suggest that such hazards would pose approximately 20 times the risk of the radiological occupational hazards (Blomeke et al. 1980).

Other factors, such as seismic activity, fires, or severe meteorologic conditions, could lead to abnormal conditions. No such factors or their ensuing impacts, however, have been identified as warranting detailed environmental analysis for the transmutation facilities.

Natural System Impacts

Transmutation activity would involve handling several chemicals posing a potential health hazard. These chemicals would represent a threat to the natural environment surrounding fuel handling and processing facilities, as well as to the interconnecting transportation networks. Individual impact scenarios have not been postulated, but it can be assumed that there would be a risk of nonradiological impact associated with use of these chemicals not unlike that experienced by certain chemical process industries today.

Other nonradiological ecosystem impacts would result from construction, operation, and maintenance activities. Such impacts cannot be fully addressed except for a specific site. In general, potential impact would be similar to that of a comparably sized industrial operation. Reductions in the quantities of natural vegetation, an increase in runoff, and elimination of certain habitats are types of impacts that would be expected from such a facility. Although similar to impacts described for the baseline case of a fuel reprocessing operation that includes a mined geologic repository, the transmutation impacts would be greater because additional facilities and increased transportation would be involved.

Socioeconomic Impacts

Socioeconomic impacts associated with the transmutation alternative would occur primarily as a result of construction, operation, and transportation activities. Implementation of this alternative would involve a major construction force of over 3,000 individuals. Employment needs during operation would diminish to approximately 350 individuals per year for the FRP-WTF and 250 for the FFP-WTE (Smith and Davis 1980). These activities would also support increased transportation employment.

Compared to the baseline case of reprocessing without partitioning, operational employment levels for transmutation would increase substantially at the reprocessing and MOX fuel fabrication centers. Estimated work force increases are 35 and 80 percent at reprocessing and fuel fabrication facilities, respectively. Estimated socioeconomic impacts of such facilities are only conjectural at this point and specific impacts of hypothetical communities and groups are not included in this discussion.

Aesthetic Impacts

No data exist suggesting that aesthetic concerns from facilities required for transmutation activities would be greater than those associated with the reprocessing without partitioning. Neither the appearance or noise levels produced from the additional partitioning facilities should vary significantly from the baseline fuel reprocessing and preparation facilities.

Resource Consumption

Fuel and raw materials used in construction, as well as the chemicals and fuel required during operations and subsequent transportation activities, would be the most important resources used in the partitioning and transmutation process. For construction activities, a range of energy sources would be used in hardware fabrication and in actual construction operations. Other building materials such as steel, sand, and gravel typically used in major construction activities would also be consumed.

The reprocessing and partitioning process would also require quantities of chemicals, including nitric acid, hydrofluoric acid, hexanitrate acid, and several solvents. These chemicals would react with the waste material to form secondary wastes, as well as the desired end products.

Additional land would be required for this alternative. Facilities at the reprocessing plant should occupy 70 ha (172 acres) (Smith and Davis 1980) compared with 36 ha (90 acres) at present (DOE 1979c), and at the fuel fabrication plant 24 ha (59 acres) (Smith and Davis 1980) compared with 3 ha (8 acres) at present (DOE 1979c). Such a facility would normally process approximately 400 MTHM/year. In addition to the acreage occupied by each facility, large "restricted" areas would have to be established. Because of the conceptual nature of these facilities and the many possible ways they might be laid out, there are no specific estimates of the total size of restricted areas. At a minimum, the combined reprocessing and

waste treatment facility would require a 2400 ha (6000-acre) restricted area while the fuel fabrication plant would require a 4000-ha (10,000-acre) restricted area. These figures are based on estimates for the reprocessing and fuel fabrication plants without waste treatment facilities (DOE 1979c).

International and Domestic Legal and Institutional Considerations

The primary institutional concern associated with implementation of a transmutation process would be the compatibility between such a system and existing power reactors. Specifically, the use of commercial power reactors as transmutation devices might result in significant fissile penalties, reactor peaking problems, reduced reactor availability, shielding requirements for fresh fuel, increased operating costs, and the need for significantly more enriched ^{235}U as a driver fuel. Consequently, technological improvements in transmutation processes or an evaluation of the institutional framework surrounding establishment of new nuclear plant operating standards is needed before the transmutation alternative can be implemented.

Finally, it must be recognized that the partitioning and transmutation processes include intensive reprocessing of nuclear waste material and plutonium recycle. Adoption of the transmutation alternative therefore, would be inconsistent with this nation's current policy regarding reprocessing.

6.1.7.5 Potential Impacts Over the Long Term (Postemplacement)

Successful implementation of the transmutation process would reduce the long-term hazards associated with waste material. In fact, effective transmutation would virtually eliminate concerns with actinides and their daughters. Although the potential long-term benefits would be significant, there are long-term uncertainties and problems that must be weighed against them.

Potential Events

For this option, TRU-depleted wastes are assumed to be sent to a mined geologic repository. Therefore, events leading to potential problems over the long term for this option would be the same as those associated with the mined geologic repository (see Section 5.6). A major difference exists in impacts, however, because transmutation wastes would not be as toxic in the long term (beyond 1,000 years).

Potential Impacts

Impacts over the long term would be expected to be less severe than those anticipated with reprocessing only, since the waste placed in the repository would be partitioned and transmuted to reduce its toxicity. An important exception to this would occur following early termination of the transmutation cycle. Such termination can actually increase the radiotoxicity of the wastes, as mentioned earlier (Croff et al. 1977).

Results of a long-term risk comparison (Logan et al. 1980) between a reference (no transmutation) and a transmutation fuel cycle indicate that:

- Cs-137 and Sr-90 would dominate the health effects during the first few hundred years for both fuel cycles.
- After a few hundred years and for several tens of thousands of years thereafter, the most significant nuclides for the reference fuel cycle would include a generous mix of actinides and their daughters at a significantly reduced activity level. Transmutation would strongly reduce the effects during this period.
- During later years, two nuclides, Tc-99 and I-129, which are released by leaching, would completely dominate all other nuclide contributions. Because these nuclides are not removed through transmutation, the results show no benefit during these later years.

Long-term health effects have been integrated over 1 million years to determine the long-term probabilistic (expected) risk (Blomeke et al. 1980 and Logan et al. 1980). The long-term risk was found to be controlled to a very large extent by the contributions from Tc-99 and I-129, which constitute about 99 percent of the integrated risk. This is because (1) the slow leach incident dominates the long-term probabilistic risk since it was assumed to have a much higher probability of occurrence than a volcanic or meteor incident and (2) only those nuclides that sorb poorly or not at all (i.e., iodine, technetium, carbon) migrate through the geosphere quickly enough to reach the biosphere within 1 million years. Therefore, transmutation of actinides would have its most substantial value if an unlikely event occurs. For example, the probability of a volcanic incident is only one in 100 billion, but if it should occur, the radioactive material could enter the biosphere very rapidly.

Looking at the issue described above in another way, it is noteworthy that catastrophic events occurring beyond 100 years following emplacement would not cause significant radiologic health effects if transmutation were employed.

6.1.7.6 Cost Analysis

The cost of utilizing transmutation to modify the radionuclide composition of waste would be added to the cost of disposal associated with remaining modified waste. However, modification of the waste's radionuclide content has the potential to alleviate some of the disposal requirements and reduce these costs. Such costs have not been developed at this time.

Costs have been developed for a fuel cycle including actinide transmutation utilizing commercial light water reactors as the transmutation device. These were compared with the costs of a mixed-oxide fuel cycle (Alexander and Croff 1980). This study indicated cost increase of about 3 percent for nuclear generated electricity if actinide transmutation were utilized for disposal purposes.

The significant cost differentials were associated with the requirement of specialized partitioning facilities and hardware. The continued recycle of actinides into the fuel cycle would increase the neutron activity within the fuel material about tenfold for spent fuel and

more than 100 times for fresh fuel. These increases must be taken into account by increased shielding and by use of remote operations and maintenance when designing fuel cycle facilities. Reprocessing costs would increase by an estimated 5 percent, fuel fabrication costs would double, and transportation costs would nearly triple (Smith and Davis 1980).

The following cost estimates are for only the specialized partitioning facilities collocated with their respective mixed-oxide fuel fabrication facility and spent fuel reprocessing facility. The fuel fabrication plant has a throughput of 660 MTHM per year and the reprocessing plant a throughput of 2,000 MTHM per year.

Capital Costs

The partitioning process buildings are first-of-a-kind facilities that, in several instances, include process operations that have not advanced beyond laboratory test and evaluation. Therefore, considerable judgment was used in the development of the capital costs shown in Table 6.1.27.

Operating Costs

Estimated operating costs are shown in Table 6.1.28. Labor cost estimates are based on an average salary of \$20,000 per year for management, engineering, and supervision and \$14,500 per year for operators, maintenance personnel, guards, laboratory technicians, and clerical personnel.

TABLE 6.1.27. Capital Costs For Partitioning Facilities
(Millions of 1978 Dollars)
(Smith and Davis 1978)

	Colocated With Reprocessing Plant			Colocated With Fuel Fabrication Plant		
	Material	Labor	Total	Material	Labor	Total
Land Improvements	1.3	1.2	2.5	1.0	.9	1.9
Process Facilities	200.0	127.0	327.0	73.1	46.9	120.0
Tunnel and Piping	5.8	10.6	16.4	4.9	9.2	14.1
Support Facilities	13.0	5.7	18.7	12.2	4.6	16.8
Subtotal	220.	145	365.	91	62	153
Field Indirects and S/C's OH&P			145			62
Subtotal			510			215
Engineering & Design			143			60
Subtotal			653			275
Contingency			228			96
Total			881			371

TABLE 6.1.28. Operating Costs For Partitioning Facilities
(Millions of 1980 Dollars)

	<u>Colocated With Reprocessing Plant</u>	<u>Colocated With Fuel Fabrication Plant</u>
Process Chemicals	16.0	1.4
Utilities	6.2	2.2
Labor	8.2	5.8
Equipment Replacement	3.8	1.0
Property Tax and Insurance	26.0	11.1
NRC License and Inspection	<u>0.2</u>	<u>0.2</u>
Total	60.4	21.7

Decommissioning

Decommissioning costs associated with the partitioning facilities were estimated to be 12 percent of the capital costs for the partitioning facilities, i.e., \$105 million for the facility colocated with the reprocessing plant or \$45 million for the facility colocated with the fuel fabrication plant.

6.1.7.7. Safeguard Requirements

The transmutation concept depends on processing of the spent fuel elements and the recycle of transmutable materials. The extra processing and transportation, and the availability of sensitive materials at all points in the back end of the fuel cycle would increase the opportunity for diversion of these materials. In addition, because of the necessity to process and recycle material eight or nine times to ensure full transmutation, the annual throughput of sensitive materials would greatly increase. Material accountability would also be more difficult because of the large quantities and high irradiation levels. Safeguards of recycled plutonium would be simplified because of the higher concentration of ^{238}Pu . Also, recycled actinides containing ^{252}Cf and ^{245}Cm would require shielding from neutrons that should simplify safeguard requirements. Furthermore, because geologic disposal would be required on the same scale as discussed in Chapter 5, all the safeguard requirements described there would also be required for a fuel cycle using transmutation. See Section 4.10 for additional discussion of predisposal operation safeguard requirements.

6.1.8 Space Disposal

6.1.8.1 Concept Summary

Space disposal offers the option of permanently removing part of the nuclear wastes from the Earth's environment. In this concept, HLW would be formed into a cermet matrix and packaged in special flight containers for insertion into a solar orbit, where it would remain for at least 1 million years. NASA has studied several space disposal options since the early 1970s. A reference concept using an uprated Space Shuttle has emerged and is considered in detail here.

The Space Shuttle would carry the waste package to a low-earth orbit. A transfer vehicle would then separate from the Shuttle to place the waste package and another propulsion stage into an earth escape trajectory. The transfer vehicle would return to the Shuttle while the remaining rocket stage inserted the waste into a solar orbit.

The space disposal option appears feasible for selected long-lived waste fractions, or even for the total amount of high-level waste that will be produced. The remaining TRU wastes would require some terrestrial disposal option, such as mined geological repositories in the continental U.S. Space disposal of unprocessed fuel rods does not appear economically feasible or practical because of the large number of flights involved.

Space disposal was considered for its potential to reduce long-term environmental impacts and human health effects for a given quantity and type of waste compared with alternative terrestrial disposal options. Because of the characteristics of the space disposal concept, which removes the waste package from the biosphere, it is highly unlikely that physical forces would cause the radioisotopes to migrate toward the Earth. Consequently, for a package properly placed in orbit, there would be no long-term risk or surveillance problem as in terrestrial alternatives. However, the risk and consequence of launch pad accident and low earth orbit failure must be compared to the risk of breach of deep geologic repositories.

6.1.8.2 System and Facility Descriptions

System Options

The reference concept and system for the initial space disposal of nuclear waste has been developed from a number of options available at each step from the reactor to ultimate space disposal. These options are summarized in Figure 6.1.22 (Battelle 1980), which indicates currently preferred options chosen for the DOE/NASA concept, primary alternatives, secondary alternatives, and options that are no longer considered viable. The bases for selection of options for the reference concept (those blocked off) are detailed in various sources listed in Appendix M.

Waste-Type Compatibility

As noted, space disposal of unprocessed spent fuel rods would be impractical because an excessive number of launches would be required. This would result in high energy re-

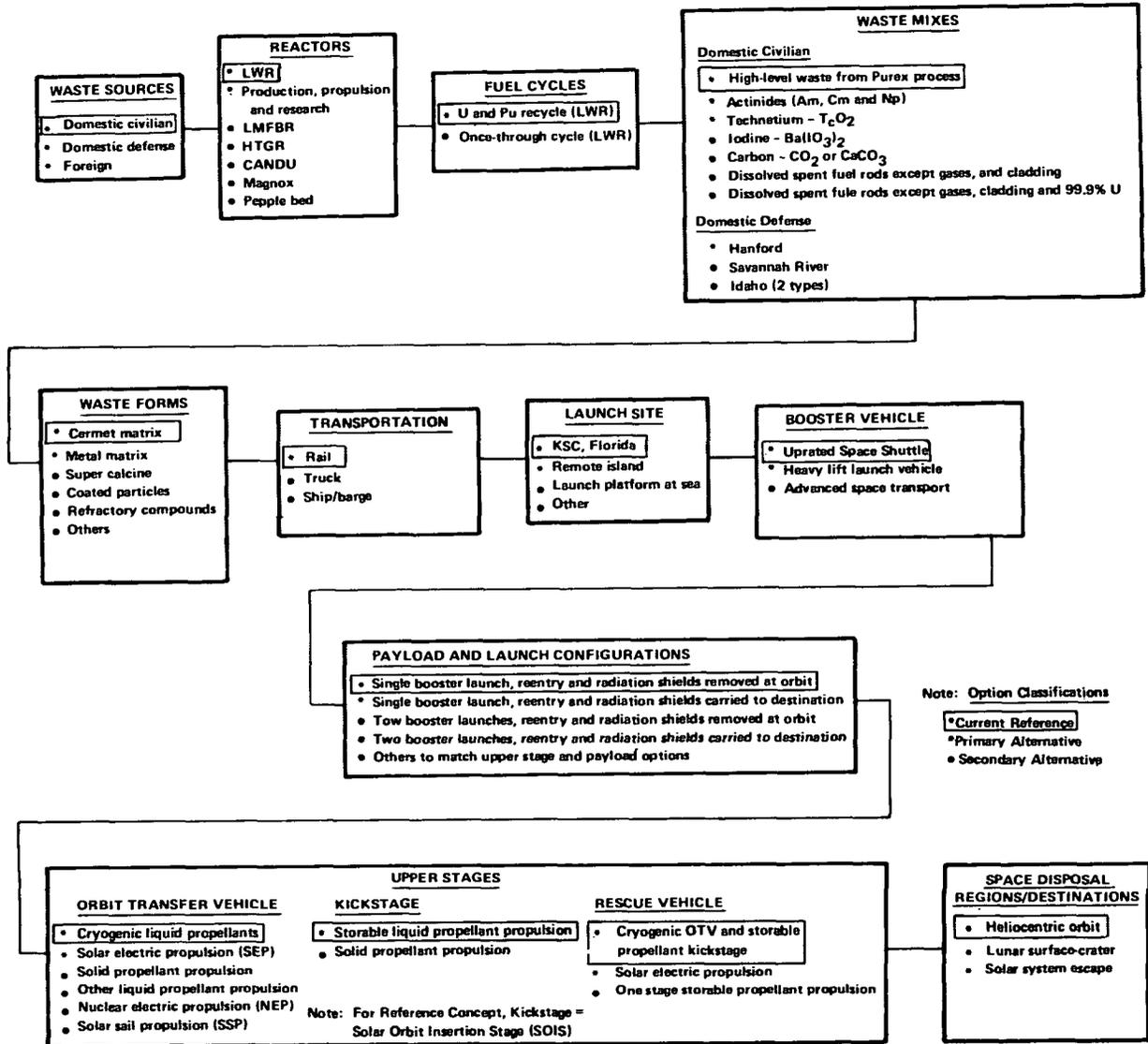


FIGURE 6.1.22. Major Options for Space Disposal of Nuclear Waste

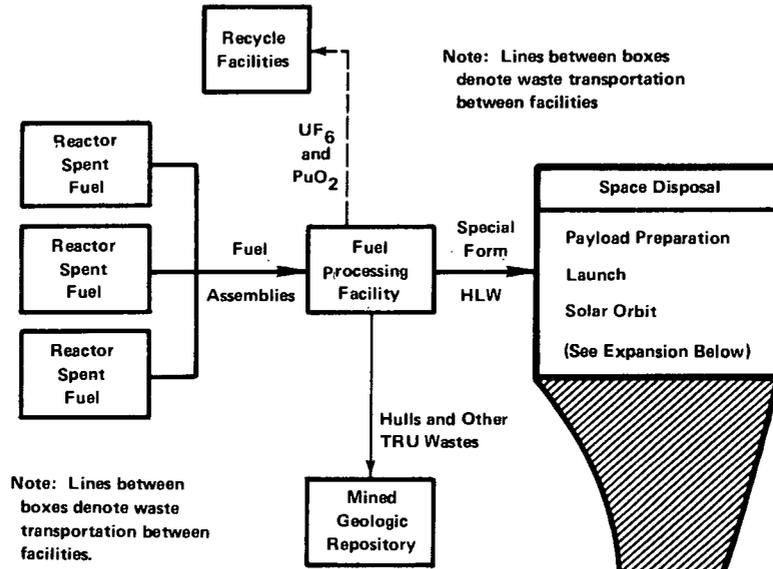
quirements, high costs, and probably increased environmental impacts (see Section 6.1.8.4). Thus, some form of waste separation would be required. For HLW, the option appears to be feasible, on the basis of the much lower number of Space Shuttle flights that would be required (approximately one launch per week to dispose of HLW from 5000 MT of heavy metal resulting from operations of approximately 170 GWe nuclear capacity). It is also possible that the space option would be used to rid the Earth of smaller quantities of radioactive wastes that pose special hazards for long-term terrestrial disposal. The disposal of selected isotopes would require chemical partitioning, with its high costs and secondary waste streams. Remotely handled and contact-handled TRU wastes from the recycle options would require geologic disposal.

Waste-System Description

The concept for space disposal of nuclear waste described here is the current DOE/NASA reference concept as reflected by the preferred options in Figure 6.1.22. To place the space disposal concept into perspective from a total system viewpoint, Figure 6.1.23 shows the waste management system, emphasizing the location and process flow details of the space disposal alternative within the total system. Two points are apparent from this figure: (1) chemical processing would definitely be required for space disposal of waste, and (2) the mined geologic repository would be part of the total system. The following discussion briefly summarizes the mission profile from the standpoint of waste-type compatibility, prelaunch activities, and orbital operations. Battelle (1980) presented a more detailed discussion of this profile and various element definitions and requirements.

Prelaunch Activities. The prelaunch activities would include nuclear waste processing and payload fabrication, ground transportation of waste, on-site payload preparation, and final staging operations.

Typically, spent fuel rods from domestic power plants would be transported to the waste processing and payload fabrication site in conventional shipping casks (see Chapter 4). A high-level waste stream containing fission products and actinides, including several tenths of a percent of the original plutonium and uranium, would result from the uranium and plutonium recovery process. This waste would be formed into a "cermet" matrix (Aaron et al. 1979) (an abbreviation for ceramic particles uniformly dispersed within a metallic phase), which has been shown to have superior properties compared with other potential waste forms for space disposal (Battelle 1980). The waste would then be fabricated into an unshielded 5000-kg sphere. Within a remote shielded cell, this waste payload would be loaded into a container, which would be closed, sealed, inspected, decontaminated, and packaged into a flight-weight gamma radiation shield assembly. During these operations and subsequent interim storage at the processing site, the waste package would be cooled by an auxiliary cooling system.



FUEL CYCLE DIAGRAM - SPACE DISPOSAL

HLW From Fuel Processing Facility

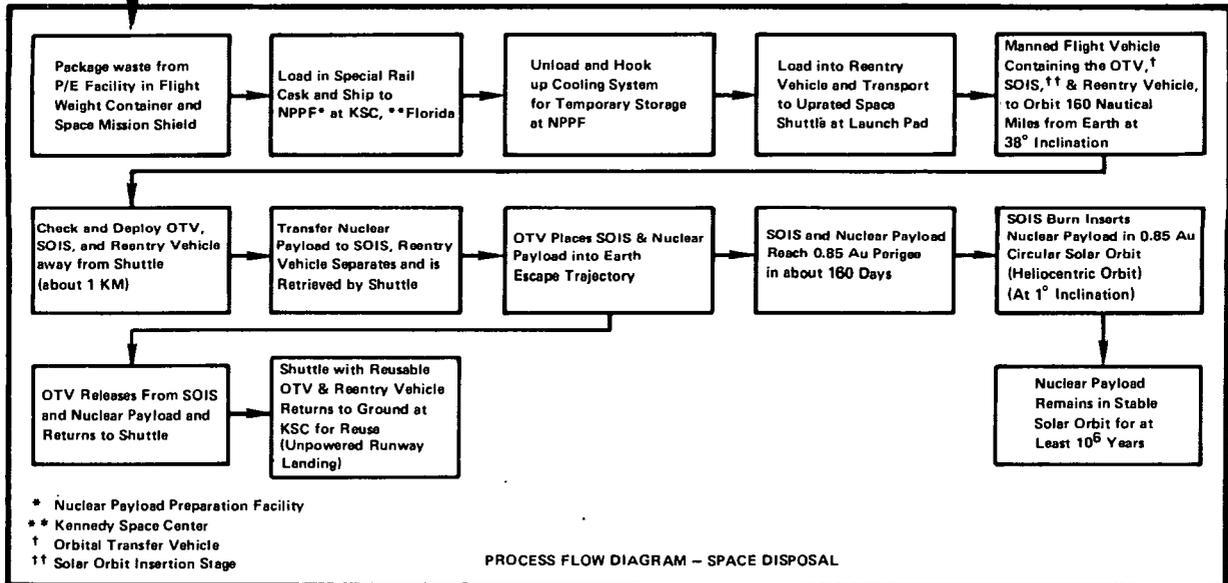


FIGURE 6.1.23. Waste Management System--Space Disposal

The shielded waste container would be loaded into a ground transportation shipping cask. This cask would provide additional radiation shielding, as well as thermal and impact protection for the waste container to comply with NRC/DOT shipping regulations. It would be transported to the launch site on a special rail car and be stored in a nuclear payload preparation facility with provision for additional shielding and thermal control. The waste containers would be monitored and inspected during storage.

For launch, the shielded waste form would be integrated with:

- A reentry vehicle, which would protect and structurally support the waste in the Space Shuttle orbiter cargo bay
- A solar orbit insertion stage (SOIS), which would place the waste payload into its final solar orbit
- An orbit transfer vehicle (OTV), which would take the waste from low Earth orbit into a solar orbit transfer trajectory.

Pre-launch checkout would include verification of the payload and the payload-to-orbiter interface systems. Typically, propellant would be loaded in the preparation facility to minimize the hazard of propellant loading while the payload was in the Shuttle cargo bay on the launch pad.

From the preparation facility, a special-purpose transporter would take the payload to the launch pad, where special equipment would position and install it in the Shuttle cargo bay.

Orbital Operations. The orbital operations for this concept would include launching into earth orbit, transfer from there to a solar orbit, and finally rounding out the solar orbit. (see Figure 6.1.24). The Uprated Space Shuttle, designed to carry a 45,000 kg (99,000 lb) payload, would be launched into a low Earth orbit (300 km). The launch would avoid early land overflight of populated land masses. The liquid rocket booster engines and the external tank would be jettisoned before the orbit is reached.

During suborbital portions of the flight, the Orbiter would be able to command shutdown of all engines and either return to the launch site or ditch in the ocean. From 5 to 6 minutes after launch, the Orbiter could abort by going once around the Earth and then returning to land. After 6 minutes, the Orbiter has the on-board thrust capability to abort directly to a sustained earth orbit. If a Shuttle malfunction exceeded the abort capability, the nuclear payload with the reentry vehicle would automatically eject and make its own reentry. It would be designed to survive a land or water impact.

Once in orbit, the loaded reentry vehicle would be automatically latched to the SOIS and, with the OTV, would automatically deploy from the orbiter bay. At this time, the waste payload would be remotely transferred from the reentry vehicle to the SOIS payload adapter.

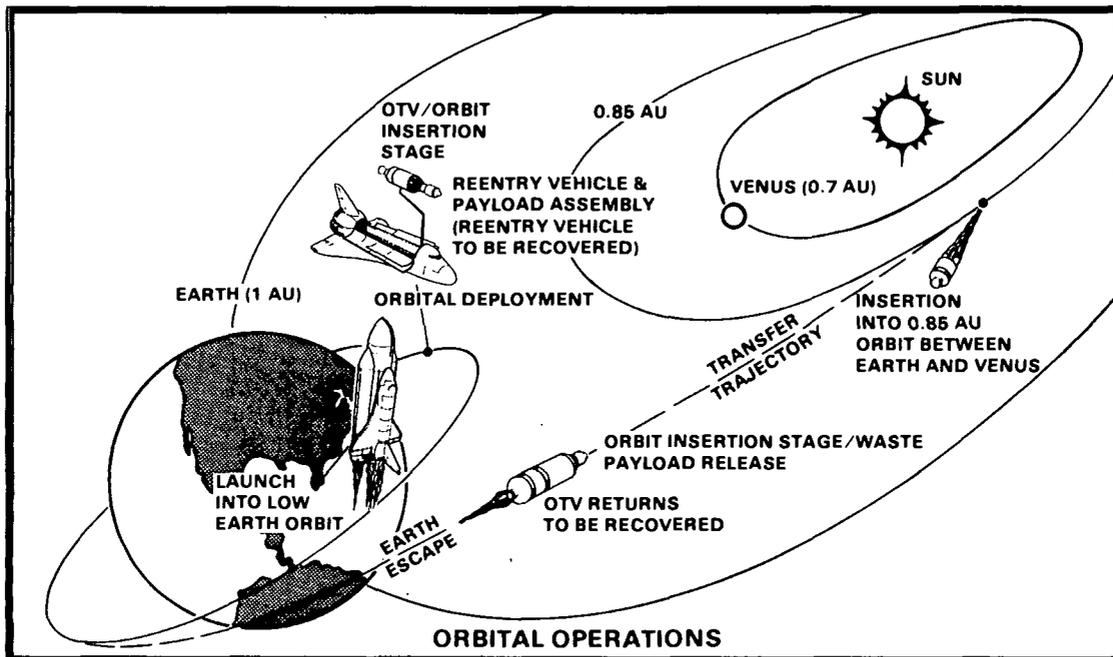


FIGURE 6.1.24. Orbital Operations

After a final systems checkout, the OTV would place the SOIS and its attached waste payload into an Earth escape trajectory. Propulsion would be controlled from the Orbiter, with backup provided by a ground control station. After propulsion, the OTV would release the SOIS/waste payload and would return to low Earth orbit for rendezvous with the Orbiter. The payload would require about 163 days to reach its perihelion at 0.85 astronomical units (A.U.) about the Sun. (One A.U. is equal to the average distance from the Earth to the Sun.) Calculations have shown that this orbit would be stable with respect to Earth and Venus for at least 1 million years.

In case of OTV ignition failure, a rescue OTV would be launched to meet and dock with the SOIS for propulsion into the escape trajectory. Safety features would be included in the design of this vehicle to prevent reentry of the unshielded payload into the Earth's atmosphere (Bechtel 1979a).

After rendezvous with the OTV, the Shuttle Orbiter would return to the launch site for refurbishment and use on a later flight. The empty reentry vehicle would also be recovered and returned with the Shuttle for reuse. The normal elapsed time from launch to return to the launch site would be 48 hours (Bechtel 1979a).

Systems for tracking the vehicles during launch, earth orbit, and the earth escape trajectory exist. There is also a system for locating and tracking the payload in deep space at any future time. However, once the proper disposal orbit had been verified, no additional tracking should be necessary.

Retrievability/Recovery. Until the waste package had been successfully disposed of in accordance with the design, retrieval or recovery capability would be necessary. A discussion of the rescue technology required for such a retrieval capability is presented in Section 6.1.8.3 below.

6.1.8.3. Status of Technical Development and R&D Needs

Present State of Development and Technological Issues

While the space option appears technically feasible, there are engineering problems that would require resolution. The Space Shuttle is currently in development and the first orbital flight is scheduled in 1981. The Space Transportation System should eventually (1990s) include a Space Shuttle with liquid rocket boosters (replacing current solid rocket boosters) and a reusable OTV. NASA has studied such vehicles extensively for future space missions and they represent a logical extension of the space transportation capability upon which to base a reference concept.

Many aspects of the space disposal system represent straightforward, applications of existing technology, e.g., use of liquid propellants and reentry vehicle design; however extensive engineering development would be required. The major technology development requirements are in design for safety, environmental impact analysis of space launches, and waste preparation. The nuclear waste payload container and reentry vehicle are only conceptually defined and additional study would be required to assure that safety and environmental requirements could be met in case of launch pad and reentry accidents. Development of a capability for deep space rendezvous and docking to correct improper orbit of a waste package would be required. The current status of development and research needs in specific areas are discussed below.

Emplacement Methods. The technology for launching both nuclear and nonnuclear payloads into space is highly developed, but the technology for putting nuclear waste in space is still in a conceptual stage. Earlier experience with space nuclear auxiliary power (SNAP) systems employing radioactive thermoelectric generators provides some experience, particularly in safety analyses, but the amounts of radioactive materials in such systems are much less than those that would be associated with waste payloads. The present DOE/NASA conceptual definition is based on technology and equipment used previously in other space missions but which would require design modifications for use in waste disposal missions. For example, the Space Shuttle power plant would need to be upgraded to increase payload capacity and thereby reduce the number of flights required. On the basis of the results obtained in the space program, considerable confidence has been gained in ability to design the necessary high-reliability systems. Procedures currently being developed to address abort contingencies for the manned Space Shuttle would be useful to mitigate adverse effects of aborts in waste launch operations.

Waste Form. The waste form would have to be a nondispersible, chemically stable solid. The composition of this waste has not been defined by the space program sponsors, but there are several possible candidate processes that might produce the proper form, as suggested in Figure 6.1.22.

The waste form should contribute to overall system safety, especially for potential accident sequences, and should also contribute to system optimization in terms of payload, economics, and materials compatibility. Desirable attributes are:

- High HLW to inert content ratio
- High thermal conductivity
- Resistance to thermal shock
- Thermochemical stability
- Toughness
- Low leachability
- Applicable to both commercial and defense wastes
- Resistance to oxidation
- Low cost
- Ease of fabrication.

Because weight would be important in the launching operation, the waste forms should also maximize the amount of waste carried at each launch (waste loading). An iron/nickel-based cermet prepared by ORNL for other disposal options appears suitable, but would require further development.

Waste Package. The reference waste package would consist of the spherical waste form surrounded by a metal cladding, a gamma shield, a steel honeycomb structure (for impact), insulation (for reentry), a graphite shield (for reentry), and the reentry vehicle itself, which would contain the waste during launch and Earth orbit in case of accident. Only conceptual definitions have been developed.

Waste Partitioning. Certain space option alternative concepts would be enhanced if specific isotopes were removed from the waste, e.g., strontium or cesium. Alternatively, space disposal might be more appropriate for certain species, e.g., iodine, technetium, the actinides, or all three. Technology development would be needed to provide these partitioning options.

Facilities. The size, capacity, and functional requirements of the nuclear payload preparation facility are not defined. Major design tasks remain before this facility could be developed.

Rescue Technology. Remote automated rendezvous and docking capabilities would probably be required for space disposal of radioactive waste. The HLW payload would require technology development to provide recovery capabilities for payloads in deep space, especially for uncontrollable and/or tumbling payloads. Also, it might be necessary to develop new technology for deep ocean recovery of aborted or reentrant payloads. Deep ocean recovery has been demonstrated on several recent projects, but any new, special capabilities to handle HLW payloads would need to be defined. Special equipment to recover reentrant payloads that touch down on land might also be required, although the technological challenge would probably not be as great.

R&D Requirements

In the final analysis, R&D needs would depend on the space disposal mission selected. The R&D requirements for this program would span the spectrum from systems definition conceptual studies through generic technology development (e.g., waste form) to engineering developments of facilities and hardware (e.g., the payload preparation facility and tailored space vehicles). These latter aspects would be deferred until the space disposal mission is better defined.

Thus, initial R&D would need to cover the following elements for concept definition and evaluations, listed approximately in sequential order.

- Perform trade-off and risk analysis studies to select the mix of radionuclides for space disposal
- Assess technology availability of waste processing and waste partitioning options
- Develop waste form criteria and options for space disposal
- Define facilities and ground transportation systems requiring R&D
- Define waste payload systems and containment requirements
- Define and select flight support systems for the space disposal option (e.g., shielding)
- Complete conceptual definition of unique launch site systems
- Assess advanced launch systems under development for space disposal applicability
- Define possible systems for transferring nuclear waste from Earth orbit and recovering failed payloads
- Characterize possible space destinations and missions
- Assess unique safety and environmental aspects of the space mission (e.g., launch pad fires and explosions affecting the waste package).

These conceptual studies would set the requirements for future R&D programs, if warranted. Other applicable ongoing R&D projects, e.g., concept definition of metal matrix waste forms and advanced launch system definition, would be pursued concurrently.

Implementation Time

With the space disposal mission currently in the concept definition and evaluation phase, meaningful predictions of the initial operational date are not possible. However, the present DOE/NASA concept depends on the availability of an OTV and the Up-rated Space Shuttle that have not been developed. This space disposal system could be operational possibly by the year 2000. Major sequential outputs that could be derived from conceptual studies are:

- Identification of viable alternative space systems concepts
- Identification of viable nuclear waste system concepts
- Selection of preferred concepts
- Selection of baseline concept
- Completion of baseline concept definition
- Generation of development plan

Estimated Development Costs

Development costs would depend largely on the specific space option approved. Also, once that option was defined, ongoing work oriented to other Shuttle and waste disposal options could be refocused on space disposal requirements. Examples are deep space rendezvous and docking techniques and waste form technology development. This would identify the incremental Shuttle and waste isolation program costs attributable to space disposal.

Thus, funding requirements for development of the space disposal option have not been well defined. It would generally be assumed that NASA would undertake the development of the required space components and DOE would develop the waste technology if the concept was pursued. It assumed that the approach would be on an incremental basis. This work would include R&D and identification of design development requirements for nuclear waste systems and space systems for disposal, domestic/international affairs studies, and impact assessments. The studies would provide a cost basis for further programmatic decision making.

Summary

Major uncertainties, shortcomings, and advantages of the concept are summarized below:

- The concept does not permit ready corrective action.
- The concept is susceptible to single mode (launch pad) failure, unless well-engineered multiple barriers are developed to protect the waste.
- Significant technology advances and equipment development will be required.
- Waste form and package concept development are in a very preliminary stage.
- The concepts usefulness would be limited to waste from reprocessing or further limited to selected isotopes.

6.1.8.4 Impacts of Construction and Operation (Preemplacement)

A space disposal approach must consider the total integrated system risk, i.e., the risks of launching wastes into space and the risks associated with the secondary waste streams generated by waste treatment, the fraction of waste that would have to go to terrestrial disposal, and the increase in system complexity. Hence, the short-term health and environmental impacts would likely be increased, while risks associated with those residual waste forms that remained on Earth for disposal in a mined geologic repository would likely be decreased. The environmental and health impacts associated with the latter consideration are expected to be less significant than those associated with total terrestrial disposal of HLW.

In the early years of a space disposal program, certain modifications would be required at Kennedy Space Center, assuming it was selected as the launch site. At the least, this would involve construction of a payload preparation facility. If the total Space Shuttle traffic (including all space missions) saturated the capability of shuttle facilities, then modifications, or even new facilities (e.g., launch pads), would be necessary. New construction activities would be designed to have the minimum adverse effect on the area. NASA has concluded that all potential nonradiological environmental impacts foreseen during normal operation of the Space Shuttle would be localized, brief, controllable, and of minimum severity (NASA 1978). Results of an evaluation of the incremental impacts of construction of facilities to accommodate waste disposal via the Shuttle and other environmental impacts of the space disposal program are presented below (Bechtel 1979a).

Health Impacts

Normal operation of facilities are not expected to cause any significant adverse health effects from either radiological or nonradiological sources. During abnormal operations (a reentry and burnup accident) the total population radiological dose could be quite large; although the estimated average individual dose would be very small.

Radiological Impacts. Health impacts from routine operations would be related primarily to planned release of radioactive and nonradioactive materials. Impacts to man from routine operations would be derived from three of the five operational phases: predisposal treatment and packaging (reprocessing), transportation, and emplacement.

No significant adverse health effects would be expected from normal operation of reprocessing facilities (NRC 1976). Incremental effects of additional processing to partition specific nuclides are not expected to change this conclusion.

Health effects caused by terrestrial transportation would be expected to be no different for space disposal than for other waste disposal options and are assumed to be similar to those for existing containers that have been reviewed for safety and licensed by regulatory agencies.

The estimated total occupational whole-body radiation dose from space disposal (the three operational phases plus the terrestrial repository for secondary waste) is 6340 man-rem/yr

(Bechtel 1979a). (See Table 6.1.30.) Of this dose, 1000 man-rem/yr derives from Space Shuttle-related activities. The nonoccupational dose is estimated at 180 man-rem/yr, with a negligible amount attributed to the Space Shuttle program.

Accidents may be classified by their location within the sequence of operations as associated with:

- Waste treatment
- Payload fabrication
- Payload ground transportation
- Handling and launch preparation
- Launch phases (suborbital)
- Orbital operations
- Postemplacement.

Within this sequence, many possible accidents that might be called "typical industrial" accidents can be identified. These are not discussed further because they (a) are not related directly to either the nuclear or space transportation aspects, (b) have negligible environmental impact, and (c) are no more probable (and in fact may be less probable) in this activity than in any industrial activity of similar magnitude. Of primary concern here are those accidents involving radioactive material, that would lead to the release and dispersion of the radioactive material into the environment. Waste treatment, payload fabrication, payload ground transportation and handling, and launch preparation for the space disposal option would be expected to be broadly similar to the same activities as employed for terrestrial disposal options. Thus, the possible accidents and accident consequences would also be similar (subject to some variation relating to the different nuclides that might be involved). Such accidents and their consequences are treated in Chapter 4 and are not further described here.

Certain types of accidents that might occur during the launch or orbital and post-emplacment operations would impose difficult environmental conditions on the payload. They could lead to the payload coming to rest in uncontrolled areas or to the release and dispersion of some of the radioactive waste. These accident types would include:

- Explosions
- Intense fires
- High-velocity impact
- Atmospheric reentry.

The payload and other mission hardware, as well as the procedures used to carry out the various operations, would be designed to

TABLE 6.1.30 Short Term (Preemployment) Radiological Impacts For
The Space Disposal Program Normal Operation

	Whole-Body Dose, man-rem/yr	
	<u>Occupational</u>	<u>Nonoccupational</u>
Waste Processing Facility	4100	90
Transportation	210	90
Repository (Secondary Waste)	1030	Neg.
Space NPPF	70	Neg.
Transporter/Launch Pad	150	
Shuttle	<u>780</u>	
	<u>6340</u>	<u>180</u>

- Minimize the probability of events leading to severe environments
- Provide, when possible, a contingency action to remove the payload from the threatening environment
- Maximize the probability that the waste payload containment will not be violated if subjected to the environment.

Two important types of accidents, both unique to the space disposal option, are:

- A catastrophic, on- or near-pad explosion and fire of the booster launch vehicle
- A high-altitude reentry and burnup of an unprotected nuclear waste container, with subsequent conversion of a certain fraction of the payload to submicron particles of metal oxides.

Aside from immediate possible casualties and the close-in physical effects from, for example, the on-pad explosion and fire, the environmental impact of overriding significance for these events would be possible radiation exposure to the general public. Edgecombe et al. (1978) provides preliminary data on environmental conditions around catastrophic launch-pad accidents.

Short-term risks might or might not be lower than those for terrestrial disposal options. However, for the space disposal option to be implemented, they would have to be at an acceptable level. Reliability data for systems would be required before a risk assessment could be made. Reliabilities of the booster vehicle, upper stages, and safety systems envisioned for the space disposal mission have not yet been determined by NASA, but are expected to be high.

Regarding on- or near-pad accidents, no precise estimates of health effects from worst-case credible accidents can be made from present information. Nonetheless, dose commitments to the most exposed individual (80 rem/event) and to the population within 100 km of the site (4000 man-rem/ event) have been estimated for the on-pad accident (Bechtel 1979a). More work would be needed concerning the integrity of the nuclear waste container systems that would be employed for the space disposal option and the actual accident environments that would result. Additionally, the relationship between shielding and possible health effects during recovery from major accidents would require further technical study. Under accident conditions, however, the stability of the HLW is expected to reduce the consequences of any loss of containment (DOE 1979a).

In a space disposal reentry and burnup accident, the estimated average and individual dose is "quite small", yet the total population dose could be very large (e.g., about 10^7 man-rem/accident to the world population) (Bechtel 1979a).

Nonradiological Impacts. Generally, environmental impacts that would be caused by normal operations or nonradiological-type accidents from a space disposal option are not expected to be significant (NASA 1978). Potential environmental impacts related to the normal operations of space transportation systems that might be unique are discussed below.

The types of environmental health impacts that could be attributed to normal space transportation activities are:

- Gaseous and particulate emissions from rocket engines
- Noise generated during launches and landings (including sonic booms)
- Commitments of nonrecoverable resources.

These effects have been studied by NASA and an environmental impact statement has been issued (NASA 1978). To date, research has indicated there would be no significant effects to the human population from a steady launch rate of 60 shuttle flights per year.

During abnormal conditions, the major nonradiological concern appears to be whether or not large pieces of metal would reach the ground in the event of an upper stage failure. This question and others are the subject of ongoing investigations.

Natural System Impacts

Radiological and nonradiological impacts are analyzed below for the natural system.

Radiological Impacts. Environmental studies of the Barnwell Nuclear Fuel Plant (AGNS 1971, 1974; Darr and Murbach 1977) provide information concerning environmental impacts expected from normal processing of the reference waste mix. Expected environmental effects include modest heat additions to local water systems, as well as both gaseous and liquid releases of radioactive and nonradioactive materials.

In general, normal operation within regulatory limits should assure that ecosystem radiological impacts are acceptable. These conclusions are confirmed by generic studies (DOE 1979b).

The data base for environmental assessment of the space option is very preliminary at this time. Environmental assessments could be made only when the total system has been better defined. Bechtel (1979a) provides a recommended schedule for assessing ecosystem impacts from abnormal events, which, if adhered to, would make preliminary results available late in 1980.

Nonradiological Impacts. The major environmental impacts from construction of required waste treatment, payload fabrication, payload receiving, and launching facilities would be qualitatively similar to those of other construction activities. Construction impacts, in general, are related to resource commitments (land, water, and materials) and to effects on environmental quality and biotic communities from the pollutants and fugitive dust released by construction activities.

Water quality would be adversely affected by the creation of sedimentation resulting from runoff at construction sites, discharge of treated wastewaters and blowdown at reprocessing facilities, and salt pile runoff at the secondary waste repository (Bechtel 1979a).

Air quality during construction would be adversely affected as a result of fugitive dust and diesel equipment emissions, emissions from waste and employee transportation, and salt drift (Bechtel 1979a). On the basis of results of analyses performed for air quality, water quality, land quality, weather, and ecology during normal operations, no long-term or cumulative effects are predicted for the abiotic and biotic communities (NASA 1978).

Accidents related to Space Shuttle launches (without payloads) have been described elsewhere (NASA 1978) and are not expected to be environmentally significant.

Socioeconomic Impacts

Manpower estimates for construction and operation are a key variable in assessing socioeconomic impacts. Employment related to payload handling and launch is a differentiating factor between the space option and other waste disposal options.

Only preliminary data for the socioeconomic assessment of the space option are available at this time. A detailed assessment of the socioeconomic implications of the space disposal option would require more accurate employment estimates, information on the industrial sectors affected by capital expenditures, and identification of the specific geographic areas involved. Rochlin et al. (1976) provide a general discussion of the socioeconomic implications of nuclear waste disposal in space.

(a) While Kennedy Space Flight Center has already adjusted to many of the impacts mentioned below, selection of an alternative launch site would require additional impact assessment.

- Public Sector Economy. Current estimates of launch rates suggest that support of the entire space transportation system for the space disposal activity might require 25,000 to 75,000 employees. This work force represents a substantial payroll and a large number of households throughout the country that would constitute sizable demands for goods and services. The environmental impact statement for the Space Shuttle (NASA 1978) provides insight as to where money would be spent.
- Private Sector Economy. In addition to direct employment, the space disposal option would induce secondary employment, as well as major capital investment. This additional economic activity would, in turn, generate additional demands for goods and services.
- Population Size and Growth Rate/Population Composition. The size and geographic distribution of the work force levels would affect the magnitude and location of the socioeconomic impacts. The ability of local areas to meet such demands will affect the severity with which these impacts are perceived. Greater project definition and detail are necessary before these impacts can be accurately assessed.

Aesthetic Impacts

Aesthetic impacts for those aspects of the program unique to space disposal would be generally limited to noise and visual features.

Noise. Only the Orbiter reentry would produce sonic boom over populated areas. Extensive studies of sonic boom dynamics indicate that the maximum effects would be at the nuisance or annoyance level (NASA 1978).

Appearance. Visual effects are expected to be significant because of the eight-story preparation facility and a 100-m stack for the reprocessing facility. Of course, actual site selection could have a mitigating effect on these impacts (Bechtel 1979a).

Resource Consumption

Launches of space vehicles always commit certain resources that are never recovered.

Energy. Estimated total energy requirements for the space disposal program (construction plus 40-year operation), which are considered significant, are summarized below (Bechtel 1979a).

<u>Resource</u>	<u>Amount</u>
Propane, m ³	1.0 x 10 ⁷
Diesel fuel, m ³	1.5 x 10 ⁶
Gasoline, m ³	1.3 x 10 ⁵
Electricity, kWhr	5.9 x 10 ¹⁰
Propellants, MT	
Liquid hydrogen	2.7 x 10 ⁵
Liquid oxygen	3.7 x 10 ⁶
Rocket propellant	7.2 x 10 ⁵
Nitrogen tetroxide	2.4 x 10 ⁴
Monomethyl hydrazine	2.0 x 10 ⁴

Critical Resources. Estimated commitment of critical material resources required for construction plus 40 year operation (other than those required for launching) are characterized as follows (Bechtel 1979a).

<u>Resource</u>	<u>Amount</u>
Water, m ³	6.0 x 10 ⁷
Steel and Major Alloys, MT	
Carbon Steel	2.9 x 10 ⁵
Stainless Steel	3.0 x 10 ⁴
Chromium	5.0 x 10 ³
Nickel	2.0 x 10 ³
Major Nonferrous Metals MT	
Copper	3.8 x 10 ⁴
Lead	2.9 x 10 ³
Zinc	6.0 x 10 ²
Aluminum	8.3 x 10 ⁴
Concrete, m ³	1.1 x 10 ⁶
Lumber, m ³	4.0 x 10 ⁵

Land. Approximately 9000 ha (22,230 acres) of land would be required for the space disposal program. There is sufficient land capacity at the Kennedy Space Center to meet this requirement (Bechtel 1979a).

International and Domestic Legal and Institutional Considerations

The space disposal option has elements that are unique and that would have to be addressed in a comprehensive analysis of this alternative. For example, careful assignment of responsibility and accountability will have to be made among the federal agencies that would be involved in this disposal option.

The space disposal option would also present international concerns that would have to be recognized and addressed. Potential issues are:

- Risk of accidents affecting the citizens of countries that did not participate in the waste disposal decision
- Possibility of joint disposal programs with other countries
- Assignment of associated costs to various countries.

In addition to these generic international issues, there are a number of specific multi-national treaties, conventions, and agreements currently in force and subscribed to by the U.S. that bear upon the use of space for nuclear waste disposal. These include:

- "Treaty on Principles Governing the Activities of States in the Exploration and Use of Outer Space Including the Moon and Other Celestial Bodies" (1967)
- "Convention on International Liability for Damage Caused by Space Objects" (1972)
- "Agreement on the Rescue of Astronauts, the Return of Astronauts, and the Return of Objects Launched into Outer Space" (1972)
- "Convention on Damage Caused by Foreign Aircraft to Third Parties on the Surface" (1952)
- "Convention on Registration of Objects Launched into Outer Space" (1976).

This list suggests various issues that would have to be thoroughly explored in this early decision-making phase, including: (1) accident liability, (2) exclusive use of the lunar surface or other regions of outer space, and (3) international program involvement (e.g., use of the sea). These issues relate mainly to accident situations rather than routine operations.

In addition to these political and international issues, space disposal of nuclear waste would have a number of legal complexities associated with it, including liability and regulatory requirements (e.g., licensing). These concerns would be quite evident not only during, but also before and after actual implementation. Moreover, legal concerns could lengthen the time needed to implement a space disposal option.

6.1.8.5 Potential Impacts Over Long Term (Postemplacement)

Postemplacement for the space option is defined as the period of time after achievement of a stable solar orbit. Potential impacts during this period are analyzed for two different events: engineering failure and inadvertent human intrusion.

Potential Events

The possibility of sudden failure of a container in solar orbit would be extremely remote. However, if a container should rupture, for example, as a result of a meteor impact or degradation over the long term, the contents would be released and begin to spread. The physical processes by which the nuclear waste material would be dispersed in solar space include sputtering, thermal diffusion, and interactions with solar radiation and wind. Large pieces or particles of waste material would be sputtered into smaller particles, which in turn would disperse. The smallest particles, with radii less than 10^{-5} to 10^{-4} cm, would be swept out of the solar system by direct solar radiation pressure. Larger particles, those with radii up to 10^{-3} to 10^{-2} cm, would gradually lose momentum through scattering, charge exchange interactions, and collisions with energetic photons and solar wind protons. This process, called the Poynting Robertson effect, would cause these particles to begin moving in toward the sun where they would eventually be vaporized and broken down into smaller particles. Once this had occurred, the smaller particles would be swept out of the solar system by solar radiation pressure. This sweeping-out process would take an estimated 1000 to 10,000 years (Brandt 1970). NASA is currently studying this process.

The potential hazard from the isolated nuclear waste to persons on future space missions traversing the region about 0.85 A.U. is not known, but is believed to be extremely small and would be zero unless a manned trip by or to Venus were undertaken. Nuclear waste launched into an 0.85 A.U. orbit would not be recoverable for all practical purposes and the 0.85 A.U. solar orbit is far enough from the Earth and sufficiently stable that future Earth encounters would be effectively precluded (Friedlander et al. 1977).

Potential Impacts

With space disposal, waste would be isolated from the Earth for geologic time periods, in effect, permanently. Consequently, no long-term radiological or nonradiological health impacts are expected. The terrestrial component, storing only non-HLW, would therefore be minimized.

With regard to natural systems, upon retirement of waste processing fabrication and/or storage facilities (including the payload preparation facility), the land areas could be returned to other productive uses. Although details of decommissioning are not available, the various alternatives should not have a significant effect on the program. Beneficial uses of the sites by future generations would not be hindered.

6.1.8.6 Cost Analysis

Space disposal costs can be identified as follows (Bechtel 1979a):

- Waste processing/encapsulation (this may be incremental for comparisons with other alternatives)
- Ground transportation
- Launch facilities and space hardware (reusable and expendable)
- Launch operations and decommissioning
- Geologic disposal of residual nuclear wastes.

Although many of the basic space and waste technologies are understood, extrapolation to meet the requirements of the space disposal mission does not permit a valid cost estimate at this conceptual stage of the program. Initial scoping studies indicate that costs for many of these portions of the space disposal system would be similar to costs for other alternatives. The major cost difference for the space disposal alternative is attributable to the Space Shuttle operations. Capital, operating, and decommissioning costs for this incremental portion of the program are discussed briefly below.

Capital Costs

Capital costs would be incurred at Kennedy Space Center for construction of equipment dedicated to the waste disposal mission. This would include the special purpose transporter, launch pad; launch platform, and firing room. If these capital costs were recovered as

charges to DOE as a Space Shuttle user, as is contemplated for other Space Shuttle applications, they would accrue as operating costs to any DOE space disposal program. Therefore, these costs would be integrated in the per-flight charges under operating costs. One special facility not usable for other shuttle operations would be the payload preparation facility. Current estimates for this facility are \$29 million (1978 dollars). Other capital costs might accrue because of the need to allow radiation to decay in the HLW for at least 10 years prior to space disposal. Costs for such interim storage facilities have not been identified at this time.

Operating Costs

Operating costs for the space disposal alternative would be calculated on a per-flight basis, as they are for other participants in the Space Shuttle program. The per-flight cost would be approximately \$39 million in 1978 dollars.

The breakout of this estimate is:

- Up-rated Space Shuttle - \$16 million
- Orbit transfer vehicle - \$1.6 million
- Solar orbit insertion stage - \$1.6 million
- Reentry vehicles - \$5 million.

Decommissioning Costs

Decommissioning costs associated with Space Shuttle waste disposal operations would probably be limited to the facilities for waste processing and packaging, the only facilities at which contamination might be anticipated. Those decommissioning costs have been estimated at 10 percent of the initial capital costs; i.e., approximately \$3 million. Costs for decommissioning other facilities associated with the space disposal alternative are assumed to be similar to those for decommissioning facilities associated with other waste disposal alternatives.

6.1.8.7 Safeguard Requirements

Safeguards would be considered for both space disposal and the associated terrestrial disposal. For space disposal of HLW, the risk of diversion would be short-term. Once the waste had been successfully disposed of in accordance with the design, the probability of an unauthorized retrieval would be very low. Physical protection of the sensitive facilities and transportation operations would be the most effective way to deny access for the short term. Note that if this alternative were chosen for the once-through fuel cycle, despite the very high throughput required, on a purely safeguards basis it would compare favorably with many other alternatives because of the difficulty of retrieving material once it is successfully deployed. See Section 4.10 for further details on safeguards for applicable predisposal operations.

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6.2 COMPARISON OF ALTERNATIVE WASTE DISPOSAL CONCEPTS

This section provides an assessment of the nine waste management concepts discussed in Chapter 5 and Section 6.1 of this Statement.

For the reader's convenience, a brief review of each of the alternative concepts is first presented in Section 6.2.1. Next, ten assessment factors and a set of related standards of judgement are introduced. The first stage of the analysis follows, in which the concepts are screened using the standards of judgement introduced in the previous section. Concepts which remain after the screening are then compared on the basis of the assessment factors and most promising concepts identified.

6.2.1 Summary Description of Alternative Waste Disposal Concepts

This section presents brief descriptions of the nine waste management concepts considered in this comparison. Characteristics of each concept are described in more detail in Chapters 4 and 5 and Section 6.1. Technical approaches not summarized here have been advanced for certain concepts that if implemented might result in a waste management system differing from that described here. In addition, the developmental process might result in a system different than described here, especially for concepts currently in a very preliminary stage of development.

6.2.1.1 Mined Repository

In the mined repository concept, disposal of waste would be achieved by manned emplacement in mined chambers in stable geologic formations. Engineered containment would be provided by the waste form, canisters, overpacks, and sleeves. Use of a tailored backfill would provide an additional engineered barrier. Isolation and natural barriers would be provided by the host rock and surrounding geologic environment, which would be selected to provide stability, minimal hydrologic transport potential and low resource attractiveness.

A waste packaging facility would be located at the repository site where spent fuel assemblies would be individually sealed into canisters. The canisters would be incorporated into the multibarrier package and then would be placed in individual boreholes in the floor and walls of mined chambers 500 to 1,000 m deep in suitable host-rock formations. Backfill would be placed around each package following emplacement. As each chamber is ready, it would be backfilled with rock and sealed. When the repository is filled the access tunnels and shafts would be filled with appropriate materials and sealed.

All waste types referenced in Table 6.2.1 could be emplaced in the mined repository.

A reprocessing fuel cycle would produce high-level liquid waste that could be solidified to a stable waste form, packaged in canisters that are part of a multibarrier package, and emplaced in the mined repository. Transuranic waste^(a) would also be packaged and emplaced in the mined repository.

(a) Hulls, hardware, remotely handled and contact-handled TRU waste. See Table 6.2.1.

TABLE 6.2.1. Disposition of Principal Waste Products Using the Proposed Waste Disposal Concepts

	<u>Spent Fuel Assemblies</u>	<u>High-Level Liquid (Fuel Processing Waste)</u>	<u>TRU Waste(a)</u>
Mined Repository	Packaged and emplaced in mined repository.	Incorporated in immobilized solid, packaged and emplaced.	Packaged and emplaced in mined repository.
Very Deep Hole	Packaged and emplaced in deep hole repository.	Converted to immobilized solid. Packaged and emplaced in deep hole repository.	Disposal using suitable alternative technique.(b)
Rock Melt	Processed to a liquid state	Poured in rock melt repository.	Disposal using suitable alternative technique.(c)
Island Mined Repository	Packaged and emplaced in island mined repository.	Converted to immobile solid. Packaged and emplaced in island repository.	Packaged and emplaced in island mined repository.
Subseabed	Packaged and emplaced in subseabed repository.	Converted to immobile solid. Packaged and emplaced in subseabed repository.	Disposal using suitable alternative technique.(b)
Ice Sheet	Packaged and emplaced in ice sheet repository.	Converted to immobile solid. Packaged and emplaced in ice sheet repository.	Disposal using suitable alternative technique.(b)
Well Injection	Processed	Injected into geologic formations.	Disposed using suitable alternative concept.
Transmutation	Processed	Selected isotopes partitioned and transmuted to stable or shorter lived isotopes and disposed of using alternative concept.	Disposed using suitable alternative concept.
Space	Processed	Entire waste stream or selected isotopes converted to solid and emplaced in heliocentric orbit.	Disposed using suitable alternative concept.

(a) Remotely handled and contact-handled TRU wastes including dissolver solids, HEPA filters, incinerator ash wastes, failed and decommissioned equipment wastes.

(b) Could possibly be disposed of by the concept, but this is considered unlikely.

(c) Some chopped cladding and TRU wastes might be slurried into rock melt cavity subject to diluting limitations on HLW waste.

6.2.1.2 Very Deep Hole

In the very deep hole concept, disposal of high-level waste would be achieved by remote emplacement in bored shafts at depths greatly exceeding those of the mined repository. Engineered containment would be provided by the waste form, canisters, and perhaps additional barrier layers. Sorptive backfill, if used, would provide an additional engineered barrier. Isolation and natural barriers would be provided by the host rock and surrounding geologic and hydrologic environment, enhanced by the great distance to the accessible environment. The geologic and hydrologic environment would be selected to provide stability, minimal hydrologic transport potential, and low resource attractiveness.

A waste packaging facility would be located at the repository site where spent fuel assemblies would be packaged individually. The packaged fuel assemblies would be placed in rotary drilled holes as much as 10,000 m deep in crystalline rock. Holes for packages for

fuel assemblies would be approximately 48 cm in diameter. After emplacement of approximately 150 packages in the bottom 1,500 m of the hole, the hole would be sealed and filled.

A reprocessing fuel cycle would require that prior to emplacement, high-level liquid waste be converted to an immobile solid and incorporated into a multibarrier package compatible with the very-deep hole environment. TRU waste resulting from reprocessing would be disposed using other suitable disposal concepts (Table 6.2.1).

6.2.1.3 Rock Melting

In the rock melting concept, disposal of high-level and some TRU waste would be achieved by remote emplacement of liquid or slurried waste into a mined cavity. Decay heat would be allowed to melt the surrounding rock which eventually would solidify, and form a solid, relatively insoluble, rock-waste matrix. Engineered containment could be provided during the operational period by a temporary chamber lining; however, engineered barriers would not be present during the molten phase. Following solidification, the rock-waste matrix would provide quasi-engineered containment wherein the host rock and waste forms would provide suitable post-solidification properties. Isolation and natural barriers would be provided by the surrounding geologic and hydrologic environment which would be selected to provide stability, minimal hydrologic transport potential and low resource attractiveness.

Spent fuel would be converted to a slurry or dissolved at a waste processing facility located at the repository site. Plutonium and uranium could be chemically separated and sent to a mixed oxide fuel fabrication facility if a reprocessing fuel cycle were utilized. High-level waste and contact-handled TRU waste in liquid or slurry form would be piped separately to the repository. Here the waste would be injected into mined cavities approximately 20 m in diameter and 2,000 m deep. Liquid or slurried contact-handled TRU waste, supplemented with water as required, could be injected into the cavity to provide cooling. After the cavity is filled, cooling would be terminated and the injection shaft sealed. Heat from radioactive decay would melt the surrounding rock, forming a molten rock-waste mix at a temperature $\geq 1000^{\circ}\text{C}$. The mix would eventually solidify, trapping the waste within a rock matrix. Solidification should be complete in about 1,000 years.

Fuel hardware and TRU waste for which conversion to liquid or slurry is impractical would be packaged and emplaced using a suitable alternative disposal concept (Table 6.2.1).

6.2.1.4 Island Mined Repository

In the island mined repository concept, disposal of waste would be achieved by manned emplacement in mined chambers in stable geologic formations on continental islands. Engineered containment would be provided by the waste form and multibarrier package. Tailored sorptive backfill would provide an additional engineered barrier. Isolation and natural barriers would be provided by the host rock and the surrounding geologic and hydrologic environment which would be selected to provide stability, minimal hydrologic transport potential and low resource attractiveness.

Spent fuel assemblies would be packaged individually into canisters at a waste packaging facility located in the continental U.S. All canisters would be loaded into shipping casks and transported by rail to the embarkation port facility. At the port facility the waste packages would be transferred from the rail casks to ocean shipping casks which would be loaded aboard ocean-going vessels. These vessels would transport the waste to a receiving port on the U.S.-owned repository island. Waste casks would be transferred to rail or highway vehicles for shipment to the repository site. Here the canisters would be unloaded from the shipping casks, placed in multibarrier packages, and placed in individual boreholes in the floor of mined chambers at least 500 m deep in granite or basalt, located either within the fresh groundwater lens or within underlying saline groundwater. Backfill would be placed around each package following emplacement. As each chamber is ready it would be backfilled and sealed. When the repository is filled the access tunnels and shafts would be backfilled with appropriate materials and sealed.

A reprocessing fuel cycle would require high-level liquid waste to be converted into an immobile solid that would be incorporated into a multibarrier package compatible with the island geologic environment. Other wastes would be packaged and emplaced in the island repository.

6.2.1.5 Subseabed Disposal

In the subseabed disposal concept, disposal of waste would be achieved by remote emplacement in relatively thick, stable beds of sediment located in deep, quiescent, and remote regions of the oceans. Engineered multibarrier containment would be provided by the waste form, canister, and the outer body of the emplacement container. Isolation and a natural barrier would be provided by clay sediments which would be chosen for uniformity, high plasticity, low permeability, high sorption potential, long-term stability and low resource attractiveness. The ocean itself would enhance remoteness, providing protection from human intrusion. Because the ocean is part of the accessible environment it would not be considered as a barrier to waste release.

Spent fuel assemblies would be packaged individually in canisters at a waste packaging facility located in the continental U.S. Packaged fuel assemblies would be loaded into shipping casks and transported by rail to the embarkation port facility. At the port facility waste packages would be removed from the shipping casks and loaded into emplacement vehicles, probably free fall penetrometers. These would be loaded onto special oceangoing vessels and transported to the emplacement site, located in the mid-plate, mid-gyre region of the ocean with depths of 3,000 to 5,000 m. At the site the penetrometers would be released to penetrate 50 to 100 m into the clay sediment. Closing of the hole above the penetrometers might occur spontaneously or be accomplished by mechanical means and would seal the waste into the sediment. A monitoring vessel would verify satisfactory emplacement.

A reprocessing fuel cycle would produce liquid high-level waste that would be converted to an immobile solid for incorporation into a multibarrier package designed for emplacement in the sediments. TRU waste would probably require another suitable disposal concept (Table 6.2.1).

6.2.1.6 Ice Sheet Disposal

In the ice sheet disposal concept, disposal of high-level waste would be achieved by remote emplacement within a continental ice sheet. The plasticity of the ice would eventually seal the waste from the environment and subfreezing temperatures would preclude hydrologic transport except possibly at the conditions encountered at the ice-rock interface. Engineered multibarrier containment would be provided by the waste form and canisters and possibly overpacks. Isolation and a natural barrier would be provided by the ice mass. The geographic location of the repository and the inclement weather of continental ice sheets would contribute to the remoteness of the repository and decrease the possibility of human intrusion.

Spent fuel assemblies would be packaged individually in canisters at a waste processing facility located in the continental U.S. Packaged fuel assemblies would be loaded into shipping casks and transported by rail to the embarkation port facility. At the port facility waste packages would be transferred from rail casks to ocean-shipping casks which would be loaded aboard ocean-going vessels. These vessels would transport the waste to a receiving port at the ice margin. Here the waste packages in shipping casks, would be transferred to tracked vehicles for transport to the repository, located some distance inland. At the repository site the waste packages would be removed from the transport casks, placed into pilot holes drilled 50 to 100 m into the ice and tethered to anchor plates with 200 to 500 m cables or allowed to melt freely into the ice. Heat from radioactive decay would melt the ice and the package would sink into the ice sheet, reaching its final position in six to eighteen months. The pilot holes would be sealed by filling with water which would subsequently freeze. Refreezing of water above the package as it progressed downward would complete sealing of the emplacement holes.

A reprocessing fuel cycle would produce liquid high-level waste that would be converted to an immobile solid compatible with the ice environment. This solidified waste would be packaged and emplaced in the ice sheet repository. TRU waste would probably be disposed using an alternative disposal concept (Table 6.2.1).

6.2.1.7 Well Injection

In the well injection disposal concept, disposal of high-level waste would be achieved by remote emplacement of liquid or slurried waste into stable geologic formations capped by an impermeable boundary layer. A degree of engineered containment would be supplied by the waste form if a grout were used but would not be present during the injection phase. Isolation and natural barriers would be provided by the host rock and the surrounding geologic and hydrologic environment which would be selected for its stability, minimum hydrologic transport potential, high sorption potential and low resource attractiveness.

A waste processing facility would be located at the repository site where spent fuel would be dissolved and prepared for injection, either directly as a dilute acidic liquid or as a neutralized grout. The prepared waste would be transferred by piping to the injection well field. Dilute acid waste, if used, would be injected into porous sandstone having shale caprock at depths of approximately 1,000 m. Neutralized grout would be injected into a shale formation having natural or induced fractures at depths of approximately 500 m. TRU waste would require an alternative disposal concept.

Liquid high-level waste resulting from a reprocessing fuel cycle would be transferred directly to the waste preparation facility, colocated with the reprocessing plant. TRU waste would be packaged and emplaced using an alternative disposal concept (Table 6.2.1).

6.2.1.8 Transmutation

Transmutation would function as an ancillary waste treatment process for the conversion of selected long-lived waste isotopes to shorter-lived isotopes potentially reducing the time during which repository integrity must be maintained. The process would be operated in conjunction with a waste management system using a suitable alternative disposal concept for disposal of radioactive waste, including transmutation products (Table 6.2.1). Because transmutation is a waste treatment process and not a disposal alternative, it cannot be assessed in terms of containment, barriers and remoteness in the same manner as these terms are applied to repositories.

At a processing plant spent fuel would be dissolved and uranium and plutonium separated for recycle. Reprocessing wastes would be transferred to an adjacent partitioning facility where long-lived waste isotopes would be partitioned from the reprocessing waste stream. The residual waste streams, stripped of long-lived isotopes, would be processed for disposal using a suitable disposal concept. The isotopes selected for transmutation would be combined with recovered plutonium and uranium and shipped to a MOX-FFP.

At the fuel fabrication plant the plutonium-uranium-waste isotope mixture would be fabricated into MOX fuel assemblies following addition of sufficient enriched uranium to achieve the desired end-of-cycle reactivity. TRU waste from the fuel fabrication plant would be sent to a colocated waste purification facility for recovery of waste actinides. Recovered actinides would be returned to the fuel fabrication facility for incorporation into MOX fuel; the residual waste would be processed for disposal using a suitable alternative waste disposal concept (Table 6.2.1).

The MOX fuel, containing the waste isotopes for transmutation, would be shipped in shielded casks to power reactors where a portion of the waste isotopes would be transmuted to stable or shorter-lived isotopes. Transmuted isotopes would be partitioned for disposal during the subsequent reprocessing cycle. Repeated cycles would be required to achieve nearly complete transmutation of the long-lived isotopes.

Implementation of transmutation as an actinide waste treatment process requires that spent fuel be reprocessed to recover the actinides and that the actinides be recycled for transmutation, mandating a reprocessing-type fuel cycle.

6.2.1.9 Space

In the space disposal concept, disposal of selected waste products would be achieved by insertion of waste packages into a stable solar orbit approximately half-way between the orbits of Earth and Venus. Engineered containment would be provided by the waste form and its engineered package. Isolation would be provided by the remoteness of the orbit from Earth and the stability of the orbit. An additional impediment to return of waste would be provided by inclining the orbit to the ecliptic.

Spent fuel would be chopped and dissolved at a processing facility. Plutonium and uranium would be chemically separated and sent to a MOX-FFP if a reprocessing fuel cycle were utilized. Waste products for which space disposal is intended would be partitioned from the waste stream and transferred to an adjacent waste preparation facility. High-level and contact-handled TRU waste not destined for space disposal would be processed for disposal using a suitable alternative disposal concept (Table 6.2.1). Alternatively, the entire liquid high-level waste stream, including uranium and plutonium constituents, could be transferred to the waste preparation facility for space disposal.

At the waste preparation facility, the waste would be incorporated into a solid ceramic-metal composite ("cermet") which would be formed into a payload of suitable shape and size. The payload would be packed into a radiation shield and this assembly loaded into a shipping cask for transport to the nuclear payload preparation facility near the launch site.

At the nuclear payload preparation facility, the shielded waste assembly would be removed from the shipping cask and loaded into a reentry vehicle. A special transporter would then take the assembly to the launch site, where it would be positioned in the space shuttle cargo bay with an orbit transfer vehicle and a solar orbit insertion stage.

The space shuttle would be launched into earth orbit where the reentry vehicle-payload assembly would be deployed from the cargo bay. The shielded waste assembly would then be removed from the reentry vehicle and attached to the solar orbit insertion stage, which would be latched to the orbit transfer vehicle. The orbit transfer vehicle would propel the solar orbit insertion stage into an earth escape trajectory, release the solar orbit insertion stage and return to earth orbit for recovery. The solar orbit insertion stage and the waste would continue and the waste would ultimately be inserted into a stable solar orbit at 0.85 astronomical units. The space shuttle would return to earth carrying the reentry and orbit transfer vehicles.

6.2.1.10 Summary

The relationships of the nine disposal concepts to the waste products of the two primary fuel cycles have been summarized in Table 6.2.1. Products of the once-through fuel cycle include spent fuel assemblies with probably a small stream of contact-handled TRU waste resulting from fuel element failures. Five of the disposal concepts could dispose of these products directly. However, rock melt, well injection, transmutation and space disposal would require processing the spent fuel to liquid or slurry form with the result that

the spectrum of waste products characteristic of the reprocessing fuel cycle is generated. This includes liquid high-level waste, fuel hulls and hardware, and a substantial quantity of remotely handled and contact-handled TRU waste. It should be noted that the reprocessing fuel cycle will likely require an alternative disposal facility (probably a mined repository) for the high volume TRU wastes for all concepts except the island repository; mined repositories; and, perhaps, the subseabed.

6.2.2 Assessment Factors and Standards of Judgement

Ten assessment factors have been selected to facilitate comparison of the proposed waste management concepts. These factors are discussed in Subsections 6.2.2.1 through 6.2.2.10. Associated with certain of these factors are standards of judgement. The standards of judgement are applied in Section 6.2.3 to reduce the nine proposed waste management concepts to a subset of candidate concepts with greatest potential for adequate performance. Concepts in this subset are then compared in Section 6.2.4 on the basis of the ten assessment factors. The ten assessment factors are listed in Table 6.2.2 below; the assessment factors are underlined. The standards of judgement appear as bullets in Table 6.2.3 and are grouped under the (underlined) assessment factors.

TABLE 6.2.2. Assessment Factors

Radiological Effects

- operational period
- post-operational period

Non-Radiological Environmental Effects

- health effects
- socio-economic effects
- aesthetic effects
- ecosystem effects

Current Status of Development

- availability of technology
- availability of performance assessment methodologies

Conformance with Federal Law and International Agreements

Independence from Future Development of the Nuclear Industry

- industry size
- fuel cycles
- reactor design

Cost of Development and Operation

Potential for Corrective or Mitigating Action

Long-Term Maintenance and Surveillance Requirements

Resource Consumption

Equity of Risk

TABLE 6.2.3. Standards of Judgement

Radiological Effects

- A concept should comply with radiological standards established for other fuel cycle facilities.
- Containment should be maintained during the period dominated by fission product decay.
- Waste should be isolated from the accessible environment for a minimum of 10,000 years.

Non-Radiological Environmental Effects

No standards were advanced for this factor.

Current Status of Development

- The concept should be amenable to development within a reasonable period of time such that implementation is not left to future generations.
- Implementation of a concept should not require scientific breakthroughs.
- Capabilities for assessing the performance of a concept must be available prior to committing major R&D programs to its development.

Conformance with Federal Law and International Agreements

No standards were advanced for this factor.

Independence from Future Development of the Nuclear Industry

- Implementation of a concept should not be dependent upon the size of the nuclear industry.
- Concepts should be independent of fuel cycle issues.
- Concepts should be independent of reactor design issues.

Cost of Development and Operation

No standards were advanced for this factor.

Potential for Corrective or Mitigating Action

- Concepts should allow corrective action to be taken in case of failure of a system to perform as designed.

Long-Term Maintenance and Surveillance Requirements

- Reliance should not be placed on maintenance or surveillance for extended times following termination of the operational period.

Resource Consumption

No standards were advanced for this factor.

Equity of Risk

No standards were advanced for this factor.

6.2.2.1 Radiological Effects

A central objective of the nuclear waste management program is to limit radiation dose to both the public and to operating personnel to acceptably low levels. Two time periods are of interest. One is the operational period involving waste treatment, transportation, and emplacement and the second is the post operational period following termination of repository operations.

A useful measure of radiological effects during the operational period is radiation exposure resulting from emplacement of a quantity of waste derived from the generation of a

unit of electrical power by nuclear means. Unfortunately, the current state of development of many of the concepts does not permit computation of this measure. Therefore, this analysis will rely upon relative comparison, using processing and transportation requirements as secondary indicators of potential radiation dose during the operational period.

A reasonable minimum level of radiological performance during the operating period is that risks shall not be greater than those allowed for other nuclear fuel cycle facilities. This suggests a standard that appropriate regulatory requirements established for other fuel cycle facilities be met.

Objectives 1 and 2 of the proposed DOE Waste Management Performance Objectives (Table 6.2.4) are intended to provide standards related to the radiological performance of waste management concepts during the post-emplacement period. Objective 1 requires that waste containment within the immediate vicinity of initial placement should be virtually complete during the period when radiation and thermal output are dominated by fission product decay. Objective 2 requires a standard of reasonable assurance that wastes will be isolated from the environment for a period of at least 10,000 years with no prediction of significant decrease beyond that time. Both standards were adopted for this analysis (Table 6.2.3).

TABLE 6.2.4. Proposed DOE Waste Management Performance Objectives(a)

1. Waste containment within the immediate vicinity of initial placement should be virtually complete during the period when radiation and thermal output are dominated by fission product decay. Any loss of containment should be a gradual process which results in very small fractional waste inventory release rates extending over very long release times, i.e., catastrophic losses of containment should not occur.
2. Disposal systems should provide reasonable assurance that wastes will be isolated from the accessible environment for a period of at least 10,000 years with no prediction of significant decreases in isolation beyond that time.
3. Risks during the operating phase of waste disposal systems should not be greater than those allowed for other nuclear fuel cycle facilities. Appropriate regulatory requirements established for other fuel cycle facilities of a like nature should be met.
4. The environmental impacts associated with waste disposal systems should be mitigated to the extent reasonably achievable.
5. The waste disposal system design and the analytical methods used to develop and demonstrate system effectiveness should be sufficiently conservative to compensate for residual design, operational, and long-term predictive uncertainties of potential importance to system effectiveness, and should provide reasonable assurance that regulatory standards will be met.
6. Waste disposal systems selected for implementation should be based upon a level of technology that can be implemented within a reasonable period of time, should not depend upon scientific breakthroughs, should be able to be assessed with current capabilities, and should not require active maintenance or surveillance for unreasonable times into the future.
7. Waste disposal concepts selected for implementation should be independent of the size of the nuclear industry and of the resolution of specific fuel cycle or reactor design issues and should be compatible with national policies.

(a) DOE/NE-0007--Statement of Position of the United States Department of Energy in the Matter of Proposed Rulemaking on the Storage and Disposal of Nuclear Waste.

Non-Radiological Environmental Effects

Non-radiological environmental effects considered to be of potential significance in the comparison of waste management concepts include health effects from non-radiological causes, socioeconomic effects, aesthetic effects, and effects on ecosystems.

Health effects from non-radiological causes include injuries and deaths occurring to both occupational workers and to the general public from routine operations and from accidental conditions.

Socioeconomic effects include impacts on the well-being of communities in the vicinity of waste management facilities.

Potential aesthetic effects include noise, odor and impacts on visual resources.

Both natural and managed ecosystems would be affected by waste management operations. Potential impacts include those on ecosystem productivity, stability, and diversity.

No standards of judgement have been advanced for non-radiological environmental effects, although all concepts would be expected to comply with standards established by responsible Federal and state regulatory agencies. The proposed DOE Performance Objective 4 asserts the importance of minimizing non-radiation-related environmental effects.

6.2.2.2 Status of Development

This factor is intended to assess the waste management concepts on the basis of the maturity of the concepts. Two issues are of concern: 1) availability of technology required to implement the concept, including that required for site characterization, repository development, waste treatment, handling, emplacement, and monitoring; and, 2) ability to predict performance of the waste management system. A third issue, cost of research and development, is considered under the factor of cost.

Three standards of judgement relating to status of development can be derived from the proposed DOE Performance Objective 6. First the technology must be implemented within a reasonable period of time where "reasonable period of time" implies that those currently responsible can complete the major part of implementing a concept and not pass an unresolved problem on to future generations. Consequently, Objective 6 also states that scientific breakthroughs should not be required to permit implementation of a concept. Further capabilities for assessing the performance of any particular waste management concept must be available at the time that a decision is made to place emphasis on the development of any particular concept.

6.2.2.3 Conformance with Federal Law and International Agreements

The purpose of this factor is to identify and compare potential conflicts with Federal legislation and international treaties, conventions, and understandings to which this nation is a party that would prevent implementation of a proposed option. The DOE proposed Performance Objective 7 states that waste management systems "should be compatible with national

policies" suggesting that concepts might be rejected because of potential policy conflicts. Because Federal legislation and international agreements can be amended for reasonable cause, this condition will not be used as a standard, but its consideration provides insight into the difficulty of implementation. Any waste management concept, if implemented, would be required to comply with applicable laws and regulations.

6.2.2.4 Independence from Future Development of the Nuclear Industry

Implementing a nuclear waste management system is a large scale, costly, and long-term effort. Concepts selected for priority development should be independent of the future development of the nuclear industry including industry size, fuel cycles, and reactor designs.

Three standards of judgement derived from DOE Performance Objective 7 are related to this factor: 1) waste disposal concepts selected for implementation should be independent of the size of the nuclear industry, 2) independent of specific fuel cycles and 3) independent of reactor design issues.

6.2.2.5 Cost of Development and Operation

The purpose of this factor is to compare concepts on the basis of estimated costs for research and development (presumably to be borne by the Federal government but recovered from the utilities through fees charged for disposal) and on costs of implementation and operation (borne by utilities and included in their rate bases). No standards have been established for cost.

6.2.2.6 Potential for Corrective Action

The probability of system failure can be reduced to low levels by careful design, thorough assessment of performance and provision of redundant systems. However, as with any engineered system, probability of failure cannot be entirely eliminated, with the result that there will remain a probability (although very low) that the system may not perform as expected. Thus the ability to detect and correct failure or to mitigate its consequences would be a desirable property of the concept selected for implementation. The desirability of corrective action capability is implied by DOE Performance Objective 5 which suggests that corrective action capabilities should be provided to compensate for residual uncertainties in system performance. Thus the importance of corrective action capability should be assessed with consideration of residual uncertainties in system performance.

The proposed NRC Technical Standards for Regulating Geologic Disposal of High-Level Radioactive Waste require retrievability, a form of corrective action, to be maintained for 50 years following termination of waste emplacement operations (Proposed 10 CFR 60.111(a)(3)). No standards were established for corrective action potential given the dissimilar characteristics of certain of the waste management options.

6.2.2.7 Long-Term Maintenance and Surveillance Requirements

Future generations cannot reasonably be expected to assume a burden of maintaining and monitoring the nuclear wastes of present generations. Thus a desirable assessment factor for waste management concepts is that they require minimal maintenance or monitoring following decommissioning. The Environmental Protection Agency has included in its draft standards for waste management a stipulation that surveillance and maintenance should not be relied upon for a period exceeding 100 years after termination of active disposal operations (43 Fed. Register, Section 221, November 1978). A more general performance standard was adopted for this analysis that reliance should not be placed on maintenance and surveillance for extended times following termination of the operational period.

6.2.2.8 Resource Consumption

Any waste management option would require the consumption of certain resources including energy, critical nonfuel materials, and land. Certain materials which are important to a waste management option may be in short supply, potentially producing market disruptions or increased dependence on uncertain supplies. Potentially critical materials are listed in Table 6.2.5. It is important that no waste isolation approach use an unreasonable amount of any critical resource, but no specific standard is advanced.

TABLE 6.2.5. Potentially Critical Materials(a)

Aluminum	Cobalt	Nickel	Water
Antimony	Columbium	Platinum	Natural Gas
Asbestos	Graphite	Potash	Electricity
Bismuth	Iodine	Quartz (crystals)	Coal
Cesium	Manganese	Tantalum	Petroleum-Derived Fuels
Chromium	Mica	Tin	Other Fuels

(a) The nonfuel minerals of this group are considered to be "major problems from the national viewpoint" by the U.S. Bureau of Mines because of U.S. low-grade resource or reserve inadequacy to
Year 2000

6.2.2.9 Equity of Risk

Although the responsibility for disposal of high level radioactive waste belongs to the Federal government, the implementation of a specific solution will require cooperation with the state and local governments, and with the general public. A few localities will be required to accept and service the facilities for disposal of waste that was created in providing service and benefits to a very broad segment of the country's population. Consequently, the implementation of a disposal method will have to be judged against the equity of risk by the political subdivision involved.

6.2.3 Application of Performance Standards

The nine proposed waste disposal concepts are examined in this section with respect to the performance standards advanced in Table 6.2.3. Results of this judgement are tabulated in Table 6.2.6. The subset of concepts meeting these standards are subjected to more detailed comparative analysis in Section 6.2.4.

6.2.3.1 A Concept Should Comply with Radiological Standards Established for Other Fuel Cycle Facilities

The unique characteristics of several of the proposed waste disposal concepts set them quite apart in design and operation from any existing fuel cycle facility. Thus, although it is appropriate to evaluate the concepts on current dose, risk and emission standards, it may be inappropriate to apply regulations relating to the means of achieving these standards. It is not evident, based on available information, that any of the nine proposed concepts would necessarily fail to comply with dose, risk and emission standards; though it is likely that the radiological releases would vary among the concepts.

6.2.3.2 Containment Should be Maintained During the Period Dominated by Fission Product Decay

"Containment" is defined in the NRC proposed technical criteria for regulating geologic disposal of high-level radioactive waste as "keeping radioactive waste within a designated boundary" (Proposed 10 CFR Part 60). Because of inherent differences among the concepts, the following definitions of containment are used for this assessment:

- Mined Repository--Waste is contained within the waste package (Proposed 10 CFR Part 60.)
- Very Deep Hole
Island Mined Repository
Ice Sheet Disposal } Waste is contained within the package.
- Rock Melt--Waste is contained within the rock-waste matrix, and in the intended location.
- Subseabed Disposal--Waste is contained within the package (penetrometer case or overpack).
- Well Injection--Dilute Acid: Waste is contained within the intended region of the host formation
Shale-Grout: Waste is contained within the grout matrix, and in the intended region of the host formation.
- Transmutation--None, the containment concept is not applicable.
- Space--Waste is contained within its package within the predetermined heliocentric orbit.

Based on these definitions of containment, engineering judgement indicates that containment for several hundred years could likely be achieved using the mined repository, very

deep hole, island mined repository, subseabed, ice sheet, and space disposal concepts. Uncertainties, however, are associated with the very deep hole concept depending on depth of emplacement and associated conditions of temperature and pressure to which the package is exposed.

Because the rock melt concept does not provide a system of engineered barriers, and because of the elevated temperatures, it appears likely that heated water vapor or liquid could contact, leach and transport waste from the as yet unsolidified rock-waste matrix of the rock melt concept during the initial 1000-year post-operational period.

Because the well injection concept does not provide a series of engineered barriers, one thousand year containment could not be assured with either of the well injection proposals. Diffusion of dilute acid injected waste into fractures and discontinuities of formations adjacent to the host formation could be expected.

In conclusion, it appears probable that containment of emplaced waste, as defined, could be maintained through the period dominated by fission product decay for all concepts except rock melt and well injection. The containment concept does not apply to transmutation.

6.2.3.3 Waste Should Be Isolated from the Accessible Environment for a Minimum of 10,000 Years

Ten thousand years has been proposed as a time period during which the radiotoxicity of properly treated waste would decay to levels comparable with the natural uranium ore bodies from which the materials were originally derived (Voss 1980). "Isolated" is interpreted as "segregation of the waste from the accessible environment within acceptable limits" (Proposed 10 CFR Part 60) where the accessible environment includes the atmosphere, the land surface, surface waters, oceans and presently used aquifers (Proposed, 10 CFR Part 60, 40 CFR Part 146). "Acceptable limits" has been generally interpreted to include releases resulting in dose rates within the normal variation of naturally occurring radiation dose rates (DOE 1980).

Analysis to date of the mined repository concept suggests no reason to believe that acceptable isolation could not be maintained by the geologic environment for a 10,000-year period, with the possible exception of very low probability catastrophic accident situations. The probability of these occurring is estimated to be small. Similarly, it appears quite possible that the very deep hole concept could maintain acceptable waste isolation over the required period if such depths are successfully isolated from ground water.

Maintenance of waste package containment cannot be assumed for the 10,000-year period for the mined repository, very deep hole, island mined repository, subseabed disposal and ice sheet disposal concepts. Package failure would expose the waste form to a saturated hydrologic environment for the subseabed and island disposal concepts and acceptable isolation would be dependent upon stability of the hydrologic environment and the sorptive properties of the host material and surrounding geologic environment. Available evidence indicates that acceptable isolation could be maintained using the subseabed concept. Satis-

factory performance of the island concept, while possible, is less certain because of an incomplete understanding of island hydrologic systems.

Maintenance of isolation for the requisite period under ice sheet conditions appears to be sufficiently questionable as to preclude this option from further consideration on the basis of this standard of judgement. If not tethered, the packages would descend to the ice-rock interface where the waste form packages could be pulverized by ice motion, and waste subsequently transported to the ocean by water potentially present at the interface. If tethered, ice sheet erosion or sublimation (possible within a 10,000-year period given historical climatic fluctuation) could expose waste to the surface environment.

The waste-rock matrix of the rock melt concept would potentially be exposed to severe hydrothermal alteration and leaching conditions late in the cooling phase when hot water may be present at the periphery of the rock-waste mass. This could result in transfer of waste to ground water. However if the surrounding geologic and hydrologic conditions were suitable, migration of waste to the accessible environment might be limited to acceptably low levels. On the other hand, thermomechanical disruption of the surrounding geology by the rock melt process might allow rapid transfer of contaminated ground water to surface aquifers, especially if promoted by thermal gradients from decay heat. While there is currently insufficient evidence to eliminate rock melt from further consideration on the basis of this standard of judgement, satisfactory performance appears highly uncertain. Furthermore a method for resolving this uncertainty does not appear to be available.

The host rock is the primary isolation mechanism for the shale-grout version of well injection. Assuming a suitably stable formation of adequate sorptive potential, preliminary calculations (Section 6.1.6) indicate that the likelihood of unacceptable quantities of radionuclides reaching accessible ground water is small. For dilute acid injection, assuming the site has suitable bounding formations, it also appears that there would be a low probability of unacceptable quantities of radioisotopes reaching accessible aquifers. However, prediction of acceptable long-term performance of well injection will require thorough characterization and understanding of the host formations and surrounding geology. It is highly uncertain at this time how this could be accomplished.

The transmutation concept may not require repositories providing 10,000-year isolation if all long-lived isotopes are eliminated. However, the 10,000-year isolation standard is not applicable to the transmutation process per se.

The space disposal concept appears to have most merit with respect to isolation. It has been calculated that a stable orbit would provide a minimum of 1 million years isolation.

In conclusion, it appears that all concepts with the exception of ice sheet, rock melt, and well injection have the potential of meeting the 10,000-year standard for acceptable waste isolation.

6.2.3.4 The Concept Should be Amenable to Development Within a Reasonable Period of Time Such That Implementation is Not Left to Future Generations

Necessary implementation time^(a) for the ice sheet concept is estimated to be 30 years or greater (Section 6.1.5) primarily because of the substantial uncertainties which remain to be resolved regarding ice sheet stability, structure, and dynamics and understanding of waste-ice interaction. A minimum time of 20 years is also projected for transmutation (Section 6.1.7); it is unlikely that this concept could be implemented prior to the turn of the century given the need to resolve theoretical uncertainties, and establish siting criteria; and the time required for pilot plant development, construction, and testing, and construction of commercial-scale facilities.

Development time has not been projected for the well injection concept. Although the engineering requirements for this concept do not appear difficult, requirements for improved site characterization techniques, performance assessment methods and monitoring technology appear to be formidable. However it may be possible to implement this concept within 20 years.

The remaining 20 years of this century would appear to be adequate for implementation of any of the remaining concepts, if it is assumed that very deep holes may be less than 10,000 m deep.

In summary, it appears that all concepts with the exception of ice sheet and transmutation qualify on this standard of judgement.

6.2.3.5 Implementation of a Concept Should Not Require Scientific Breakthroughs

Several concepts would require significant extension of existing technology to achieve satisfactory implementation; but none of the concepts appear to require scientific breakthroughs. Transmutation might be most efficiently accomplished in a fusion reactor, which would require a scientific breakthrough.

6.2.3.6 Capabilities for Assessing the Performance of a Concept Must Be Available Prior to Committing Major R&D Programs to Its Development

The need for substantial additional performance assessment capabilities appears to exist for all concepts. While the mined repository will require refinement of performance assessment capabilities, it is believed that this will be achieved in the near future. Manned inspection of the emplacement location is currently being proposed by the NRC. If this should be applied to all concepts, it would eliminate subseabed, very deep hole, ice sheet, well injection, space, and probably rock melt concepts.

All concepts, with the exception of transmutation, space, and subseabed require further development of remote sensing capability for assessment of the characteristics of the potential host media. In addition, the well injection and rock melt concepts would require

(a) All estimates of time assume that the concept discussed receives priority for funding.

development of methods for prediction and measurement of waste location and configuration. The lack of predictive methods for the ice sheet concept appears sufficiently intractable at this time to preclude consideration of this concept.

6.2.3.7 Implementation of a Concept Should Not Be Dependent Upon the Size of the Nuclear Industry

The rock melt, transmutation and space options appear to be potentially sensitive to the size of the nuclear industry. The reference rock melting concept would require sufficient waste product to operate at least one cavity (40,000 MTHM equivalent waste) and succeeding increments would be equally as large. The minimum size of a rock melt cavity has not been determined, however, and it is possible that smaller increments would be feasible. Transmutation would require operating reactors for the transmutation step and a sufficiently large industry to justify the investment in specialized support facilities. Space disposal, as well, would require a sizable investment in specialized hardware, needing a substantial nuclear industry to justify this investment. This, however, is an economic question and does not intrinsically disqualify space disposal from consideration.

6.2.3.8 Concepts Should Be Independent of Fuel Cycle Issues

Fuel cycles treated in this document include the once-through cycle and full uranium-plutonium recycle; however other cycles are possible. Although the uranium-only fuel cycle was discussed in the draft of this Statement, review comments indicate that this cycle is not considered reasonable by the industry or the scientific community and therefore this cycle is not considered further. Additional fuel cycle issues relate to timing of fuel cycle implementation and defense wastes.

Once-Through and Reprocessing Fuel Cycles

As summarized in Table 6.2.1, the mined repository and island mined repository concepts would be capable of accommodating all waste products of both the once-through and reprocessing fuel cycles. Various considerations suggest the use of mined repositories for bulky equipment and for the considerable volume of TRU wastes, hulls, and hardware generated by the reprocessing fuel cycle for disposal concepts that cannot accommodate these wastes.

The rock melt and well injection options could find application with either the once-through or the reprocessing fuel cycles. Fuel processing would be required for the once-through cycle.

The space disposal concept, as well, could find application to either fuel cycle, however, partitioning of the waste as well as processing of spent fuel would be required if the once-through fuel cycle were used.

Transmutation would find its most promising application with the reprocessing fuel cycle. Processing and partitioning of spent fuel and recycle in a reactor would be required and alternative disposal technology would be needed for disposal of other transmutation waste products, high-level liquid fission product waste and fuel hulls and hardware.

Timing

The timing of implementation of a waste management system could potentially affect the feasibility of the concepts because of declining decay heat generation rates or by the availability of facilities required to implement the concept. Substantial reduction of decay heat rates prior to emplacement of spent fuel or high-level waste could conceivably affect the operation of the rock melt and the ice sheet concepts; however reduction in decay heat rates over the time frames being considered for deferred fuel cycles do not appear to be great enough to materially affect operation of either of these concepts. Postponement of waste disposal operations beyond the period when light water power reactors were the dominant commercial type could impact the transmutation concept by requiring alternative transmutation devices. However, alternative devices, including fast breeder fission reactors and fusion devices, may be available and probably superior to light water reactors (Croff et al. 1980). Thus it is not felt that any concept can be dismissed on the basis of timing alone.

Summary of Fuel Cycle Issues

In summary, it appears that all of the concepts offer some potential benefit with any fuel cycle and that none should be dismissed because of sensitivity to fuel cycle issues (although the case for transmutation with a once-through fuel cycle appears to be quite marginal). Pursuit of the rock melt, well injection, transmutation or space disposal concepts with either fuel cycle would require concurrent development of one of the concepts capable of disposing of TRU waste, probably a mined repository.

6.2.3.9 Concepts Should Be Independent of Reactor Design Issues

None of the concepts appear to be especially sensitive to reactor design issues.

6.2.3.10 Implementation of a Concept Should Allow Ability to Correct or Mitigate Failure

This standard tends to favor those concepts in which wastes may be readily retrieved if observations of their actual behavior under full-scale implementation reveal previously unanticipated defects in the disposal system. Mined geologic disposal lends itself most readily to this requirement although obviously attempts at transmutation could easily be abandoned if large-scale operations failed to work.

Those concepts in which retrieval from a large-scale system would be difficult or impossible fail to meet this requirement. These concepts include space disposal, rock melt, well injection, and under certain circumstances, ice sheet disposal.

6.2.3.11 Maintenance or Surveillance Should Not Be Required for Extended Periods Following Termination of Active Repository Operations

The resolidification period of 1,000 years required of the rock melt concept would appear to require surveillance for a substantial period to verify long-term stability and satisfactory containment of the molten mass. This is seen as sufficiently contrary to this

standard of judgement as to prohibit preferred consideration of the rock melt option. The other concepts appear not to be affected by this consideration.

6.2.3.12 Summary

The performance of the nine proposed disposal concepts against the standards of judgement is summarized in Table 6.2.6. It should be emphasized that these conclusions are based largely on judgement of the authors, based in many cases on fragmentary or qualitative information. Of the nine proposed concepts, mined repository, very deep hole, island mined repository, subseabed, and space disposal have the potential for meeting all of the standards. A comparison of these five concepts is given in the next section.

6.2.4 Comparison of the Waste Disposal Concepts with Most Potential

This section compares the mined repository, island mined repository, very deep hole, subseabed and space disposal concepts on the basis of the assessment factors introduced in Section 6.2.2.

6.2.4.1 Radiological Effects

Operational Period

During the operational period, occupational exposure due to waste management would be dominated by that associated with waste processing. Transportation of TRU waste represents the greatest source of dose to the general public because of the large volume of material. Additional dose to both occupational workers and to the general public could result from accidents.

Occupational radiological effects attributable to processing operations would likely be quite similar for the mined repository, very deep hole, island mined repository, and subseabed options because the waste treatments are similar. Slightly greater occupational exposure could be expected with the very deep hole and subseabed options should it be decided to section bulky TRU-contaminated equipment for disposal by these options--an unlikely decision. Space disposal would require dissolution of spent fuel for both once-through and reprocessing fuel cycles, potentially resulting in greater radiological effects compared to the other options.

Transportation and handling requirements of spent fuel from power reactors to the waste treatment/packaging facilities would be approximately equivalent for each of the disposal concepts. The mined repository and very deep hole emplacement facilities could be colocated with the treatment/packaging facility so that no additional transportation is required. Alternately, the packaging facility could be located elsewhere. Subseabed would probably require two additional transport operations--transfer of waste packages to the embarkation port and subsequent ocean transport to the disposal site. Island repositories would require one additional movement, from the receiving port to the repository and would thus be equivalent to space disposal which would be characterized by a maximum of four major transport links for high-level waste. A smaller number of links could result from appropriate coloca-

TABLE 6.2.6. Performance of Proposed Waste Management Concepts on Ten Performance Standards

	<u>Radiological Standards</u>	<u>1,000-Year Containment</u>	<u>10,000-Year Isolation</u>	<u>Developmental Time</u>	<u>Scientific Breakthroughs</u>	<u>Predictive Capability</u>	<u>Industry Size</u>	<u>Fuel Cycles</u>	<u>Reactor Design</u>	<u>Ability to Correct or Mitigate Failure</u>	<u>Maintenance & Surveillance</u>
Mined Repository	X	X	X	X	X	X	X	X	X	X	X
Very Deep Hole	X	X	X	X	X	X	X	X	X	No	X
Rock Melt	X	No	No	X	X	X	X	X	X	No	No
Island Mined Repository	X	X	X	X	X	X	X	X	X	X	X
Subseabed	X	X	X	X	X	X	X	X	X	X	X
Ice Sheet	X	X	No	No	X	No	X	X	X	No	X
Well Injection	X	No	No	X	X	X	X	X	X	No	X
Transmutation	X	NA	NA	No	X	X	No	X	X	X	X
Space	X	X	X	X	X	X	X	X	X	No	X

X = The concept appears to have the potential to meet this standard based on available evidence.
 No = The concept does not appear to have the potential to meet this standard based on available evidence.
 NA = This standard is not applicable to this concept.

tion of facilities. The failure of a launch vehicle presents a potential single mode failure for space disposal and rapid rescue from incorrect earth orbit would likely be required to prevent public exposure.

Although, based on present evidence, any of the concepts could probably be conducted with radiation doses no greater than those currently permitted in fuel cycle facilities, substantial differences in cumulative radiation exposure might exist among the concepts. The above analysis suggests the following order of decreasing preference among concepts based on relative radiological effects during the operational period: mined repository; very deep hole; island mined repository; subseabed; space.

Post-Operational Period

Based on present evidence, any of the five concepts compared here has the potential to perform satisfactorily in the post-operational period (Section 6.2.3). However, probabilities of satisfactory performance differ and will be used as the basis of this comparison. Factors to be considered in evaluating the post-operational radiological integrity include failure of engineered containment to perform as expected, failure of natural barriers to perform as expected, compromise of repository integrity by catastrophic natural events exceeding design standards, and compromise of repository integrity by inadvertent human activity. From the standpoint of all four considerations, space disposal probably would provide the greatest certainty of satisfactory waste isolation in the post-emplacment period. In addition, the probability of satisfactory containment for several hundred years is seen as equally likely for the remaining concepts (see Section 6.2.3) although the performance of the package in the very deep hole is somewhat uncertain. Thus this discussion will focus on the prospects for longer-term isolation.

The effectiveness of natural barriers is seen to be potentially the greatest for the very deep hole concept because of the extreme depths involved. This assumes that depth alone will provide the single most effective barrier; however, uncertainties regarding the long-term integrity of the hole seal remain to be resolved. The mined repository concept relies on shaft seals as a barrier also but appears to offer greater probability of satisfactory long-term integrity due to the ability for human access during sealing operations. The possibility of disturbing the stability of the host sediment by emplaced waste might render the performance of the subseabed option less than that of mined geologic. The lack of understanding regarding behavior of island hydrologic systems under natural or waste-perturbed conditions raises significant questions as to the performance of the island mined repository in the long-term. For this reason the island mined repository concept is considered to be the least acceptable of the concepts on the basis of potential performance of natural barriers.

Of the four non-space concepts, very deep hole appears on the basis of its remote depth to offer superior protection from catastrophic natural events. Little distinction on this basis can be made between the subseabed, and mined repository concepts. Mined repositories on islands appear susceptible to catastrophic natural events associated with changes in future ocean levels.

As discussed in Section 6.2.1, efforts would be made to avoid siting repositories in areas having known or potential resource value, reducing the motivation for human intrusion. Fresh ground water can be a valuable resource in an island environment, however, and the presence of fresh water is intrinsic to the most potential island locations. Metal-bearing nodules are found--though they are scarce and of low grade--in the section of the ocean being considered for subseabed disposal. The resulting order of decreasing preference relative to prospects for inadvertent human intrusion would be space, very deep hole, mined repository, subseabed and island.

This overall analysis suggests the following order of decreasing preference relative to prospects for satisfactory radiological performance in the post-emplacment period: space; mined repository; very deep hole; subseabed; island mined repository.

6.2.4.2 Non-Radiological Environmental Effects

Health Effects

Implementation of any of the concepts would involve high-risk construction and operation activities including mining operations at sea and operations in space. Industrial accidents will undoubtedly occur; however, insufficient evidence currently exists to establish significant differences between options.

Injuries to the public could result from transportation accidents, and based on the number of transportation links inherent in each concept to which the public would be exposed (see Section 6.2.4.1), the order of decreasing preference would be the mined repository/very deep hole, island, and subseabed/space concepts. The mined repository and very deep hole concepts are essentially equivalent in this regard, as are the island and subseabed concepts.

Socioeconomic Effects

A comparative analysis of socioeconomic effects of generic disposal options is difficult because of the site specific nature of those effects. While one can assess factors such as size and number of facilities, the types of location and the size, timing and stability of the associated work force as discriminators among technology options, this is only half of the necessary information to assess impact. The other half consists of those factors associated with the area's ability to absorb the impacts. For example in times of high employment (no labor surplus) and high housing occupancy rates (no available housing) a project which requires high levels of manpower will create a serious (negative) impact. At a time when unemployment is high and housing is available, the same project would be of a positive impact.

Since these technologies involve different types of location and transportation steps, comparison against a "generic" location is not really possible. The addition of effects across several locations is not clearly a meaningful exercise since the impacts do not summate for any given community or person.

The mined repository and very deep hole disposal option would require only packaging plant and colocated repositories. Subseabed disposal would require a port facility in addition to packaging plants and the island concept would require, in addition, a receiving port and the island repository. The space disposal option would require processing, packaging, and launch facilities. An auxiliary waste disposal system for remotely handled and contact-handled TRU waste would likely be required for all concepts except mined geologic and island repositories.

In general, construction activities near small communities impact the socioeconomic structure of the community more than construction activities near large communities. Major facilities for the island geologic and subseabed disposal options would be located near the sea coast where the work force could typically be drawn from nearby communities. For the space disposal option, launch pad facilities exist and the required auxiliary facilities could be constructed at the launch site; however the waste treatment facility would also be required. The mined repository and very deep hole repositories would be located in areas of the continental United States, possibly in remote low population areas. In the case of space disposal especially there will likely be a substantial long-term increase in local employment due to the number of people required for support of launch activities. Subseabed has the same characteristics to a lesser degree, as does island disposal.

In conclusion, insufficient evidence (on a generic basis) is currently available to permit meaningful evaluation of alternative concepts on the basis of socioeconomic factors.

Aesthetic Effects

Aesthetic effects include noise, odors, and visual impacts. Analysis of aesthetic effects requires site-specific data because the effects are quite localized and dependent upon the design and siting of facilities. Because of this, characterization and comparison of aesthetic effects is not attempted in this Statement. Aesthetic effects would be an appropriate consideration in a statement considering proposed facility construction at a specific location. Items such as spoil piles from mined repositories and mud ponds from deep hole drilling could be unsightly, but the impacted area is not large.

Ecosystem Effects

Potential impacts of waste management facilities on ecosystems include effects on productivity, stability, and diversity. Evaluation of these effects at the generic level is difficult because of the sensitivity of these primary impacts to site and design characteristics which can only be addressed when considering specific installations. Consideration of such siting or design characteristics is beyond the scope of this generic statement. Thus to assess potential effects of the waste management options on ecosystems, it is necessary to look for effects inherent in the concepts under consideration.

Potential effects of the mined repository option include preemption of habitat during construction and operation of waste processing and repository facilities, potential releases of toxic waste processing chemicals to the environment and potential release of toxic spoil materials. Some preemption of habitat is unavoidable but with appropriate location and

design might well be limited to a few hundred acres of low productivity habitat. Release of toxic materials presents a potentially more severe problem. While it is predicted that release of chemicals from waste packaging facilities can be controlled to acceptable levels, control of spoils may prove difficult because of the open air storage required.

Very deep hole repositories would produce ecosystem effects similar to the mined repository option. Spoils, however, would be less bulky and presumably easier to control.

Island geologic, though technically similar to the mined repository concept, has a greater potential for ecosystem disruption because of the sensitive and unique characteristics of many island ecosystems. Assuming careful design and management of such a facility, however, the facility exclusion area might well protect or restore the integrity of the natural ecosystem as has happened to some extent at the sites such as the DOE site near Hanford, Washington. Leach of the spoil pile could significantly effect the quality of a small island ecosystem.

The potential ecological effects of the subseabed option are not known at this time. On-shore facilities are likely to be constructed near populated (and presumably ecologically disturbed) areas because of current efforts to protect what remains of natural coastline. A large area of seabed would be subject to penetrometer emplacement; however, the population and productivity of the affected region is likely to be low and relatively minor disturbance would be experienced.

Ecological effects of space disposal are likely to be modest (with the exception of those normally associated with space flight launches) in comparison to the other options. Assuming space disposal of all high-level waste, ancillary geological repository requirements would be very small compared to disposing of all waste in terrestrial repositories.

All concepts under consideration here offer the potential for satisfactory performance on the basis of non-radiological environmental effects; however, important differences in the absolute magnitude of these effects may exist. Some discrimination is possible on the basis of non-radiological health effects to the general public; however, the generic nature of the study and the early stage of development of most of the concepts provide tenuous discrimination among concepts on the basis of occupational (non-radiological) health effects and socioeconomic, aesthetic, and ecological effects. The order of decreasing preference based on available evidence regarding non-radiological environmental effects is: mined repository/very deep hole, subseabed/island, space.

6.2.4.3 Status of Development

Availability of Technology for Construction of System

There are considerable differences among the concepts with respect to the engineering development needed for implementation. Construction for the mined repository and island repository options would use well-tested existing technology, although for novel applications. The waste treatment technology required to support the mined repository concept is also well advanced, having been the focus of substantial development. Less is understood

relative to waste treatment and packaging requirements for an island mined repository, and considerable development activity might be required if the waste form and package concepts developed for mined repositories proved unsuitable for the island repository environment. The island concept would also require development of ocean transport and related transshipment facilities. Development of this equipment, however, is not viewed as particularly difficult, but largely an extension of existing technology.

The technology and methodology for siting geologic and subseabed repositories are developed to the point that they may be implemented. Space is unique in that the final location for disposition is not severely restricted by terrestrial concerns. Other options are poorly developed with respect to siting technology.

Implementation of the subseabed option, in addition to requiring development of the transshipment and ocean transport technology, would also require development of emplacement and emplacement monitoring technology, suitable waste form and packaging for the subseabed environment, and recovery technology for emplaced waste packages.

Space disposal would require development of a number of supportive technologies. Some (e.g., the space shuttle) are currently under development for other purposes and much of the remaining hardware represents extension of existing technology.

The very deep hole concept would require a significant extension of existing technology if the 10,000-m depth is required. Of the techniques available for making deep holes only rotary drilling has been used to develop wells to depths approaching those envisioned for very deep holes. Rotary drilling has been used for drilling to depths of about 9,000 m at bottom diameters of 6-1/2 inches--both shallower and of less diameter than postulated for the reference very deep hole concept. Deeper holes of larger diameter are thought possible but have not been demonstrated. It is quite possible that 10,000-meter holes will not be required by the concept. Other current limitations include casing to required depths and tensile strength of wire rope. In addition to technology related to making the very deep hole, development of a suitable waste form and packaging is required.

Availability of Technology for Adequate Performance Assessment

All of the alternative options appear to require further development of performance assessment and integrated safety and reliability analysis; however, the extent of such development is likely to be far greater with those concepts which have not received substantial attention, especially very deep hole, island mined repository, and space disposal. Fewer performance uncertainties appear to be associated with the subseabed concept; considerable research is underway on the deep ocean environment and the sediments are a homogeneous and probably fairly predictable environment. Fewest uncertainties appear to be associated with the mined repository concept largely because of the greater amount of research that has been accomplished on this concept.

The following order of decreasing preference is suggested relative to the current status of development of the concepts: mined repository; subseabed/island mined repository; space/very deep hole.

6.2.4.4 Conformance with Federal Law and International Agreements

The mined repository and very deep hole concepts could be developed without apparent conflict with Federal law or international agreements. A conflict may arise for the island disposal concept depending upon the island location. It would appear appropriate that the island be a possession of the U.S. Transport of large quantities of waste over international waters has the potential of generating adverse response.

Potential conflict of the subseabed disposal with existing law has been examined in some detail. The dumping of high-level radioactive waste is prohibited by the U.S. Marine Protection, Research and Sanctuaries Act of 1972, and therefore, would require Congressional action for implementation. The London convention of 1972, a multinational treaty on ocean disposal, addresses the dumping of contact-handled TRU and non-TRU waste. Dumping of high-level waste is prohibited; however the treaty's prohibition against dumping arguably does not extend to controlled emplacement of high-level waste into submarine geologic formations. EPA interprets the treaty as making subseabed disposal illegal.

Certain aspects of space disposal are addressed by existing treaties. The 1967 "Treaty on Principles Governing the Activities of States in the Exploration and Use of Outer Space Including the Moon and Other Celestial Bodies" prohibits waste disposal on the moon but does not rule out waste disposal in heliocentric orbit. Nations may object to the space disposal option because the waste would travel over their territory before being propelled from earth orbit. The 1972 "Convention on International Liability for Damage Caused by Space Objects" defines the responsibility for objects falling to earth on other countries. Consideration of such liability would be required.

In summary, the decreasing order of preference emerging from consideration of possible legal constraints on implementation of the five concepts is: mined repository/very deep hole; island; space; subseabed.

6.2.4.5 Independence from Future Development of the Nuclear Industry

Of the five concepts under comparison, space disposal appears to be most sensitive to the future development of the nuclear industry since it is considered that a substantial nuclear capacity will be required to justify the required investment (Section 6.2.3).

6.2.4.6 Cost of Development and Operation

Preliminary estimates of the cost of construction and operation for the mined repository, very deep hole and subseabed concepts appear in Section 6.1. These have been compiled and converted to unit costs (mills/kWh) in Table 6.2.7. Cost estimates for the island mined repository and the space disposal concept were insufficiently complete to permit reduction to a unit basis.

Of the available unit cost estimates, the very deep hole concept appears to be the most expensive with estimated costs of 3.0 mills per kilowatt-hour (1980 dollars), not a significant proportion of typical current new construction power costs (30 to 50 mills/kWh). Because these cost estimates are very preliminary and because even the most costly option

TABLE 6.2.7. Estimated costs of Various Disposal Options (1980 dollars)

	Research and Development Cost \$ millions	Pre-Disposal Cost, \$/kgHM		Repository Costs			Total Cost, mills/kWh ^(a,b,c)	
		Once-Through	Reprocessing	Construction, \$ millions	Operating, \$ millions/year	Decommissioning, \$ millions	Once-Through	Reprocessing
Mined Repository, 6,000 MTHM/yr	3,700	100	170	2,600	87	25	0.7	1.0
Very Deep Hole, 5,000 MTHM/yr	900	100	170	2,800	2,100	40	2.5	3.0 ^(d)
Island	NA ^(e)	150	190	NA	NA	NA	NA	NA
Subseabed, 5,000 MTHM/yr	NA	150	190	760	29	54	0.8	0.9
Space, per flight	NA	210	170	NA	46 ^(f)	4	NA	NA

(a) Does not include Research and Development costs.

(b) Construction and decommissioning costs amortized over 17 years @ 7%.

(c) Waste production rate is 38 MTHM/GW-year.

(d) Includes 0.2 mills per kWh for ancillary repository.

(e) NA = not available.

(f) \$ million per flight.

appears not to significantly impact the cost of electrical power, a cost comparison should not currently be assigned significant weight in this analysis. It should be noted that the cost estimates for all concepts essentially assume that no currently unanticipated questions will arise, which is probably an unlikely assumption.

6.2.4.7 Potential for Corrective or Mitigating Action

Prior to closure and sealing of access tunnels and shafts, mined repositories (including those utilized in the island disposal concept) would allow failure detection and permit retrieval of waste canisters. This system allows flexibility to future generations as to how long they might choose to leave the facilities open to inspection. Following closure, failure detection would be more difficult, although remote instrumentation could be installed for this purpose. Corrective action would be difficult (though possible) as the location of the waste would be known and access tunnels could be reopened. Detection of repository failure exemplified by unexpected concentrations of radionuclides could allow the mitigating actions of restriction of access to contaminated aquifers and other measures including evacuation of affected areas.

Complete corrective action capability for the island mined repository concept would require development of systems for locating and retrieving casks lost at sea in the case of the sinking of a transfer ship. A similar system would be required for the subseabed concept. Transponder devices would be fitted to the casks while enroute, and location and retrieval of an individual cask from the seafloor is considered feasible using existing equipment. However, loss of a ship with waste within the hull would severely complicate retrieval operations. Retrieval of emplaced canisters is considered to be feasible using existing overcoring technology, although retrieval of a large number of canisters would likely be very expensive.

Full corrective action capability for space disposal would require a deep-ocean payload retrieval system if system failure released radionuclides to the atmosphere. No corrective action would be possible. If failure of the space disposal system were to occur after achieving orbit, backup launch and orbit transfer vehicles, and some means for correction of improper orbit would be required. Each of these is under consideration as part of the space disposal concept, and if successfully developed (along with appropriate monitoring systems), would provide corrective action capability for most situations.

Corrective action with the very deep hole concept is thought possible only while the package is attached to the emplacement cable.

In summary, mined repositories appear to offer the greatest potential for corrective action. Subseabed appears also to provide reasonable potential for corrective action with the principal problem being retrieval of waste from a transport ship lost at sea. Island mined repositories present the combined difficulties and assets of the subseabed and mined repository concepts. Full corrective action potential appears to be achievable with space disposal for all situations except failure of the waste packaging system during launch or pre-orbital operations. Corrective action is thought not to be possible with the very deep

hole concept following package disengagement. The following order of decreasing preference relative to corrective action is thus suggested: mined repository; island mined repository; subseabed; space/very deep hole.

6.2.4.8 Long-Term Maintenance and Surveillance Requirements

None of the five concepts being considered here appear to require significant maintenance and surveillance activities during the post-operational period.

6.2.4.9 Resource Consumption

Preliminary estimates of selected critical resources for mined repository, very deep hole, subseabed and space disposal are provided in Table 6.2.8. Because of the very preliminary state of development of most concepts as reflected in the apparent inconsistencies among the estimates of Table 6.2.8, comparisons on the basis of these estimates would not be meaningful.

6.2.4.10 Equity of Risk

None of the concepts appear to have significant differences in this respect. Subseabed, ice sheet, island, and space disposal have the positive feature that no one must live in close proximity to the final disposal location. This creates the initial impression that the impact and risk are far less for those alternatives than for mined repositories. However a situation is established wherein the process of transportation of wastes is channeled through one location. A judgement of the equity of risk and impact resulting from the focus of transportation versus the focus of disposal is yet to be established.

6.2.5 Conclusions

Results of the comparisons on the assessment factors are depicted in Table 6.2.9 which shows the preference rankings of the five concepts (mined repository, very deep hole, subseabed, island repository, and space) on each of the assessment factors for which discrimination was found among the concepts. For each factor, the rankings of the five waste management concepts are plotted along a preference continuum, ranging from "most preferred" at the extreme left to "least preferred" at the extreme right. Concepts are clustered where no differences were observed.

6.2.5.1 Mined Repository

Examination of Table 6.2.9 supports selection of the mined repository concept as the waste disposal concept for preferred development. This concept is a "most preferred" concept on six of the seven comparisons of Table 6.2.9, ranking second on one consideration, "Radiological Effects During the Post-Operational Period." Here, the apparent length of isolation provided by space disposal results in the latter being preferred to mined repositories. An overall evaluation of the Radiological Effects attribute, however, might place

TABLE 6.2.8. Estimated Resource Commitments for Various Repositories

<u>Critical Resource</u>	<u>Mined Repository^(a,c)</u>	<u>Very Deep Hole^(b)</u>	<u>Subseabed^(b)</u>	<u>Space^(b)</u>
Aluminum, MT	220	13,000	13,000	83,000
Chromium, MT	--	14,000	14,000	5,000
Nickel, MT	--	7,500	7,500	2,000
Water, m	1,300,000	199,000,000	--	60,000,000
Natural Gas or Propane, m	11,500	10,000,000	10,000,000	10,000,000
Electricity, kWh	3,400,000,000	56,000,000,000	20,000,000,000	59,000,000,000
Petroleum-Derived Fuel, m ³	5,300,000	6,000,000	5,100,000	1,500,000
Other Fuel, MT	--	--	--	4,800,000

(a) Highest consumption construction scenarios of Tables 5.4.2 and 5.4.3 added to operational values.

(b) Highest consumption scenario indicated of Section 6.1.

(c) Island mined repository has similar commitments.

TABLE 6.2.9. Summary of Preference Rankings

	<u>Most Preferred</u> _____ →	<u>Least Preferred</u>
Radiological Effects		
Operational Period	(MR) (VDH) (IMR) (SS)	(S)
Post-Operational Period	(S) (MR) (VDH) (SS)	(IMR)
Non-Radiological Environmental Effects	(MR, VDH)	(SS, IMR) (S)
Status of Development	(MR)	(SS, IMR) (S, VDH)
Conformance with Law	(MR, VDH) (IMR) (S)	(SS)
Independence from Future Development of the Nuclear Industry	(MR, VDH, IMR, SS)	(S)
Potential for Corrective or Mitigating Action	(MR) (IMR) (SS)	(S, VDH)

KEY: MR = Mined Repository
VDH = Very Deep Hole
IMR = Island Mined Repository
SS = Subseabed
S = Space.

space disposal in an intermediate position below mined repositories because of the low ranking of space disposal on the basis of radiological effects during the operational period.

6.2.5.2 Subseabed

No clear preference emerges between the subseabed disposal concept and the island mined repository concept. However, because of significant uncertainties regarding the long-term radiological integrity provided by island geologic and hydrologic systems, subseabed appears to be superior to the island mined repository concept for continued development as an alternative to mined repository waste disposal. An additional advantage may be provided by subseabed's unique characteristics as a genuine conceptual alternative to mined repositories in comparison with island disposal, which is basically a variant (with additional uncertainties) of the mined repository concept. Uncertainties remain to be resolved concerning the long-term integrity of the emplacement media; development of transportation, emplacement and monitoring technology; resolution of potential international conflicts; and development of corrective action capabilities. Research will still be required, especially with the objective of resolving the waste isolation potential of the subseabed sediment. Should this capability be demonstrated conclusively, engineering development of the system could proceed.

6.2.5.3 Very Deep Hole

Although not possessing any clearly defined advantages over the mined repository concept on the basis of currently available evidence, the very deep hole concept ranks generally high on most of the assessment properties. Very deep hole offers potential for a high degree of geologic barrier performance in the post-operational period and some possibility of superior working conditions compared to mined repositories. A key issue is the value of manned in-situ examination of the actual placement location to understand the condition and environment into which the waste package is to be placed. Significant problems remain however, including the need for substantial development of drilling technology, improved understanding of the geologic environment at very deep hole depths, and analytical verification of the postoperational integrity of very deep hole repositories and performance of packages at the requisite temperature and pressure. Since deep hole technology is being developed for other reasons (e.g., for geopressured methane and for geothermal purposes) it is likely that increased information will be available regarding these uncertainties. An additional problem is the difficulty of providing adequate corrective action capability. Thus, the very deep hole concept, though having potentially superior characteristics to other alternatives, is also characterized by greater uncertainties. For these reasons, although continued development of the very deep hole concept as a long-term alternative to mined repositories is recommended, the priority of development is considered to be secondary to the subseabed concept. The considerations of potential problems with corrective action and the relatively unadvanced status of technology weigh heavily in this decision.

6.2.5.4 Space Disposal

The principal argument for space disposal is its promise for extraterrestrial disposal of selected radioisotopes; but substantial reservations exist concerning this concept. These include the potential radiological risk of the concept during the operational period, non-radiological health effects, potential conflicts with international law, and the difficulty of developing acceptable corrective action capabilities. Because of these conditions, priority development of space disposal as an alternative to mined repositories would appear to be unwise.

6.2.5.5 Island Disposal

The island disposal concept appears to present few advantages over the subseabed concept or the mined repository and is characterized by significant uncertainties regarding its potential for long-term isolation of waste. The principal potential advantage of island disposal is sociopolitical--it offers the possibility of a repository site remote from habitation and, thus, possibly of greater acceptability to the general public. Furthermore, the potential for international cooperation in establishing a repository at a "neutral" site might be presented by an island. Subseabed, however, offers the same advantages; thus the island concept would have merit only if the sociopolitical advantages were seen to be highly important, an appropriate island were available, and if the subseabed concept proved not to be technically acceptable. Because of these considerations, and because of great uncertainties regarding the waste isolation potential of island geology, development of this concept is not recommended.

REFERENCES FOR SECTION 6.2

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Chapter 6

ALTERNATIVE CONCEPTS FOR WASTE DISPOSAL

A number of possible alternative methods for the disposal of nuclear waste have been suggested. These concepts have been evaluated and developed to various degrees by different organizations. The status of technology is described in this section, as are advantages and disadvantages of each concept. The intent is to address the various concepts as consistently as possible to facilitate the comparison of the potential impacts of their implementation.

The alternative concepts discussed are: the very deep hole, rock melting, island repository, subseabed, ice sheet, well injection, transmutation, and space. These are all compared to the mined repository concept.

6.1 PRESENTATION/ANALYSIS OF ALTERNATIVE DISPOSAL CONCEPTS

This section presents concept descriptions and discussions of potential health and environmental impacts for eight radioactive waste disposal methods that have been suggested as alternatives to disposal in mined geologic repositories. These presentations are based on sections from the draft of this Statement, updated to incorporate current information resulting from continuing development and evaluation of alternative concepts. Information presented here is taken from available results of relevant studies. References, cited throughout the text to indicate sources of significant parameter values and statements, are listed at the end of subsection 6.1. In addition, bibliographies are provided in Appendix M to indicate other information sources for each concept. The concept descriptions are also supported by information in Chapters 3, 4, and 5 of this EIS and reference is made to those chapters as appropriate.

The discussion of each disposal concept covers the following topics:

- Concept Summary
- System and Facility Description
- Status of Technical Development and R&D Needs
- Impacts, Both Preplacement and Postplacement
- Cost Analysis
- Safeguard Requirements.

Because concept descriptions, environmental impacts, and estimated costs for each option were taken from various sources that used different basic assumptions, the information provided here for each concept is not normalized to a standard set of conditions, e.g., a common

annual throughput or a common environment. As an example, the well injection concept section presents radiological impact information extracted from a reference which addresses the impacts of intermediate level waste disposal. This is done to provide the reader with related information that may be important to the understanding of the concept. In addition, the space disposal and transmutation concepts require chemical processing of spent fuel to prepare waste for disposal or elimination. Accordingly, comparisons between these concepts and, for example, others not requiring processing would be difficult. For instance, transportation costs in the processing case could not be compared with those for disposal of spent fuel.

Four of the concepts (very deep hole, rock melt, space, and subseabed), however, were covered in a common reference and thus have a common basis. The other options are not normalized because, for example, while linear extrapolation to a higher or lower quantity of waste handled may result in a more or less conservative estimate of impacts and costs for a particular option, it may also bias any comparative analysis for or against that concept. Also, the descriptions, impacts, and costs that have been reported for some of the alternatives are incomplete because of the early stage of the alternatives' technical development.

In addition to being, in many cases, incomplete, the cost and impact data should be considered speculative. For example, the costs projected for the development of an alternative are generally based on judgment regarding the current state of technical uncertainty for the alternative. In practice, many such cost estimates do not adequately anticipate the expanded scope of activities that may result as additional uncertainties and issues are identified in attempts to resolve the original set of uncertainties. It was felt, therefore, that manipulating costs and impact information may indicate more significance than is warranted.

The disposal methods along with rates used as a basis for defining each of the alternatives, including the mined geologic repository, are:

<u>Alternative</u>	<u>Disposal Rate, MTHM/yr</u>	<u>Reference</u>
Mined Geologic Repository	6,000	Chapter 3
Very Deep Hole	5,000	Bechtel (1979a)
Rock Melt	5,000	Bechtel (1979a)
Island	Disposal rates similar to mined geologic repository. Ocean transportation similar to sub-seabed concept, see section 6.1.	Chapter 5, and Section 6.1.4
Subseabed	5,000	Bechtel (1979a)
Ice Sheet	3,000	MITRE (1979)
Well Injection	Unspecified	ORNL TM 1533, DOE (1979)
Transmutation	2,000	Blomeke et al. (1980)
Space	5,000	Bechtel (1979a)

Frequently, numbers taken from the various references are rounded to an appropriate number of significant digits in an effort to simplify this section of the document.

The general approach to each of the topical discussions used to describe the alternatives is as follows.

Concept Summary. The concept summary provided for each alternative contains a general discussion of the disposal concept, highlights significant technical aspects of the concept, and establishes a basis for specific system and facility descriptions, technology status, and environmental impact analyses.

System and Facility Description. In this section, the systems and facilities associated with a reference repository system design for each alternative disposal concept are described. Each description begins with a discussion of the fuel-cycle options reflected in the reference system design. The options and the selections made are illustrated by a standard diagram.

The waste-type compatibility for each concept is discussed, providing a basis for defining waste types that can and cannot be accepted by the disposal system. This section also indicates if the total fuel cycle involves chemical processing and if there is a need for a mined geologic repository (or other additional facility) to accept some portion of the waste.

The waste management system descriptions cover predisposal treatment and packaging (with reference to Chapter 4), surface facilities and equipment, and transportation systems. These descriptions vary substantially because of differences among the alternatives, e.g., space disposal compared to transmutation. System descriptions provide a basis for subsequent discussion of technology status and R&D requirements, potential environmental impacts, and cost analysis.

Status of Technical Development and R&D Needs. This section provides an insight into the technical status and R&D needs associated with the development of each disposal option. The discussions are based on the most current reports contained in the large body of references available for disposal options. Emphasis was placed on documents prepared by organizations that have played a definitive role in the development or evaluation of specific options.

Each disposal option is at a different stage of development ranging from ice sheet and rock melt, which are in only the early conceptual stage, to well injection, which has been used for the disposal of remotely handled waste at the Oak Ridge National Laboratory. Wide disparity in the states of development, however, should not be used to connote the degree of difficulty anticipated in deploying a particular option.

Current technological issues unique to each option are identified. These issues depend on the state of development. As knowledge is accumulated and refined on a specific concept to resolve technical issues, it may often reveal additional technological concerns to be resolved.

Specific research and development requirements ascribed to each disposal option are those contained in references provided by organizations involved in the development or evaluation of the particular disposal option. The requirements identified are based on technological issues and programmatic needs.

Estimates for implementation time and research and development costs depend on the degree of planning information available for the disposal concept. For example: no estimates

are identified for well injection because of lack of definitive program plans. Available estimates for space disposal go through concept definition and evaluation only. Estimates for ice sheet disposal, however, include all of the currently anticipated activities required to develop and implement an operational system.

Impacts. Impacts are presented on the basis of information found in the reference material. Impacts for these sections are separated into Health Effects Impacts (the human environment) and Natural System Impacts. Natural System Impacts include impacts to ecological and geologic/hydrologic systems. The term Natural System Impacts therefore includes impacts other than those to the human environment. The reader is cautioned that for those alternatives that are more advanced in their technical development, a greater number of environmental impacts are identified. Likewise, for those disposal methods that are in a preliminary stage of development, there may be other environmental impacts that have not yet been determined.

In general, the methodology followed in calculating impacts is not described, but reference is made to original material where the reader can find this information. No attempt has been made to develop a common impact assessment methodology, so the methods applied vary from study to study. For these reasons, the values presented are not always comparable on a one-to-one basis. It is believed, however, that sufficient information is provided to allow a qualitative comparison of the alternatives (see Section 6.2).

Cost Analysis. The cost analyses provide capital, operational, and decommissioning cost estimates based on information available from references authored by organizations involved in the evaluation or development of the specific disposal options. The costs are those directly attributable to the disposal mode under consideration and not on support modes such as waste preparation or routine transportation. All cost estimates are given in 1978 dollars, derived by an adjustment of 10 percent per year of estimates based on non-basis years.

The reader is cautioned about the preliminary nature of the cost estimates. Also, in many cases, due to the underdeveloped status of most of the alternatives, full cost data are not available. In such cases only referencable information is presented. No attempt is made to estimate system or component development, capital, operating or decommissioning cost where these, estimates could not be based on open literature reference. For example, in the case of the transmutation concept, a comprehensive and conclusive fuel cycle cost analysis has not been performed such that an aggregate cost estimate could be prepared. In addition, the impacts to the costs of disposal of the residual wastes from the transmutation concept are not known.

The estimates do not include transportation and waste-form preparation costs associated with the disposal method. However, unique transportation and waste-form requirements, in addition to the need for supplemental storage or disposal, are identified.

The cost analyses for very deep hole, seabed, rock melt, and space disposal are based on estimates contained in a current reference that used consistent waste disposal rates over the same time period. The available costs for the other disposal options, including the

mined geologic repository, are not normalized to the same waste disposal scenario. Cost estimates are sufficiently accurate, however, for a qualitative comparison.

Safeguard Requirements. In this section, the vulnerability of each alternative waste disposal concept to the diversion of sensitive materials or terrorist acts of sabotage is qualitatively discussed. In addition, the features unique to the alternative that enhance or detract from that vulnerability are described. For more detailed discussion of safeguards for predisposal operations the reader is referred to Section 4.10.

6.1.1 Very Deep Hole

6.1.1.1 Concept Summary

The very deep hole (VDH) concept involves the placement of nuclear waste as much as 10,000 m (32,800 ft) underground, in rock formations of high strength and low permeability. In this environment, the wastes might be effectively contained by the distance from the biosphere and the location below circulating groundwater as they decay to innocuous levels (OWI 1978 and ERDA 1978). To act as a nuclear waste repository, the host rock would have to remain sealed and structurally stable under the heat and radiation introduced by the wastes. Potential rock types for a repository of this kind include crystalline and sedimentary rocks located in areas of tectonic and seismic stability.

An immediate question concerning this concept is: "How deep is deep enough?" The required depths would place the wastes far enough below circulating ground waters that, even if a connection develops, transport of materials from the repository to the surface would take long enough to ensure that little or no radioactive material reaches the biosphere (LBL 1979). The absolute value of this depth is not yet determined.

Defining the necessary depth at a given site requires determining site-specific limits on the transport of radioactive materials to the biosphere, the site-specific hydrologic regime, and the heat-source configuration (waste packing). Available data from the literature, primarily from the oil and gas industry, show that some sedimentary rocks are porous and permeable and may contain circulating groundwater to depths in excess of 9,000 m (30,000 ft). Investigations of crystalline rock, although very limited, suggest that at much shallower depths some such rocks have relatively low porosities and permeabilities. Hence "very deep" for these crystalline rocks may mean just a few thousand meters instead of the 9,000 m or more required for sedimentary rocks. Once the required depth has been determined, the technology for making the hole to that depth and the ability of the surroundings to accept the heat source become the limiting factors. It is clear that problems of making the hole, holding it open, emplacing the waste, and sealing the hole must be considered together. Should shallow depths be determined as adequate, many of the potential problems of the very deep hole concept (e.g., drilling technology and ambient conditions at depth) would be mitigated.

The concept assumes that disposal in very deep holes would not permit retrieval of wastes. It would also provide assurance that no climatic or surface change will affect disposal.

Environmental impact considerations for the very deep hole concept are those associated with drilling a deep well or sinking a deep shaft, constructing the predisposal surface facilities, emplacing the wastes, decommissioning the facilities, and ensuring long-term containment of the wastes.

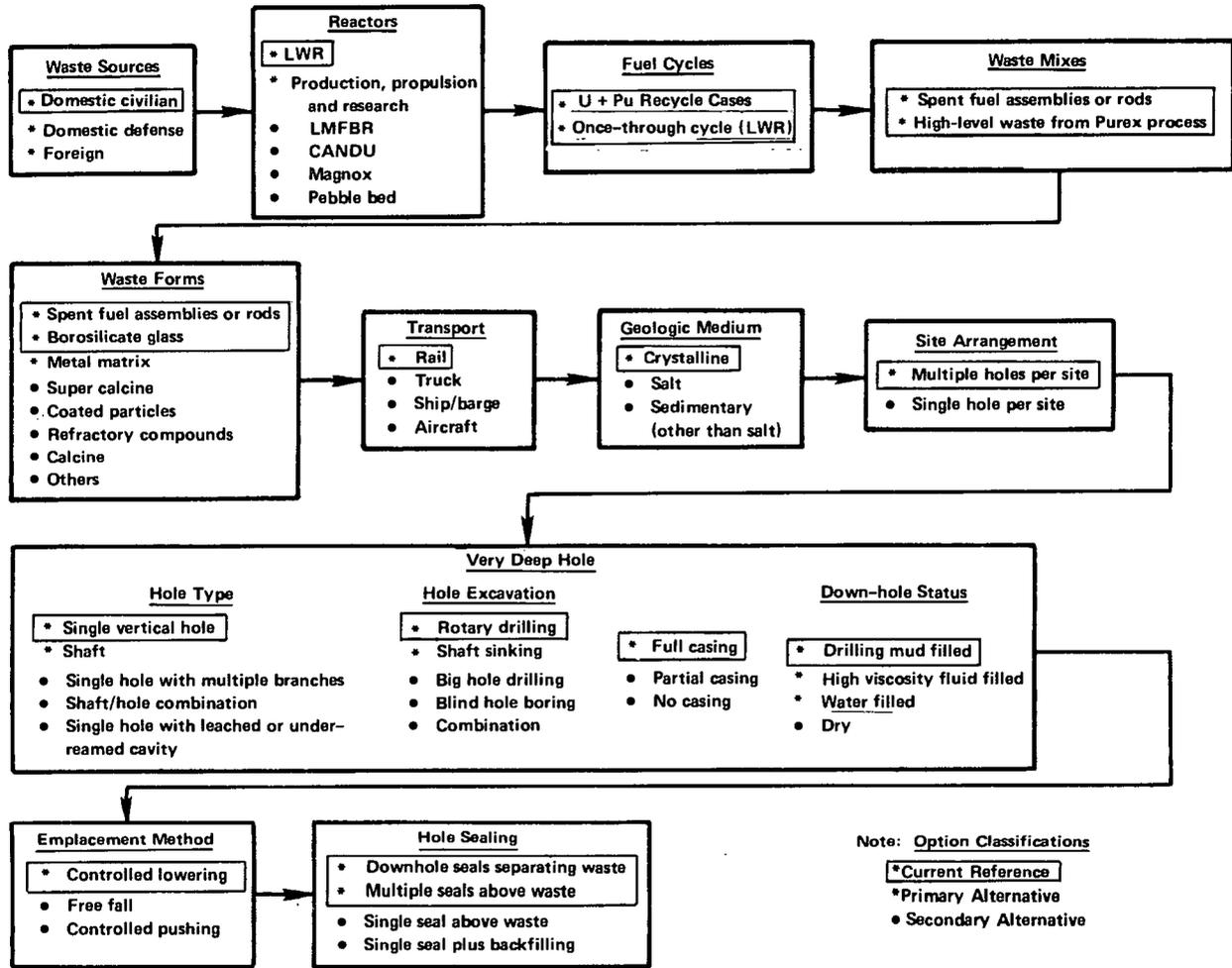
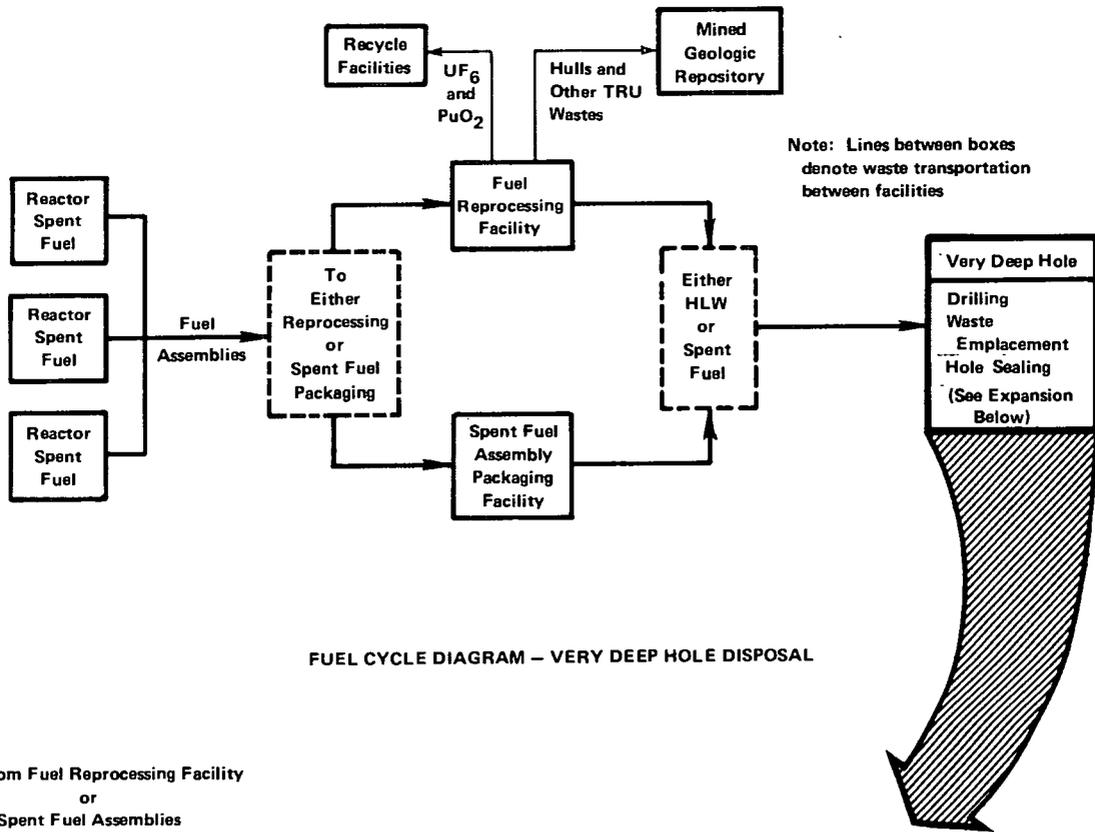
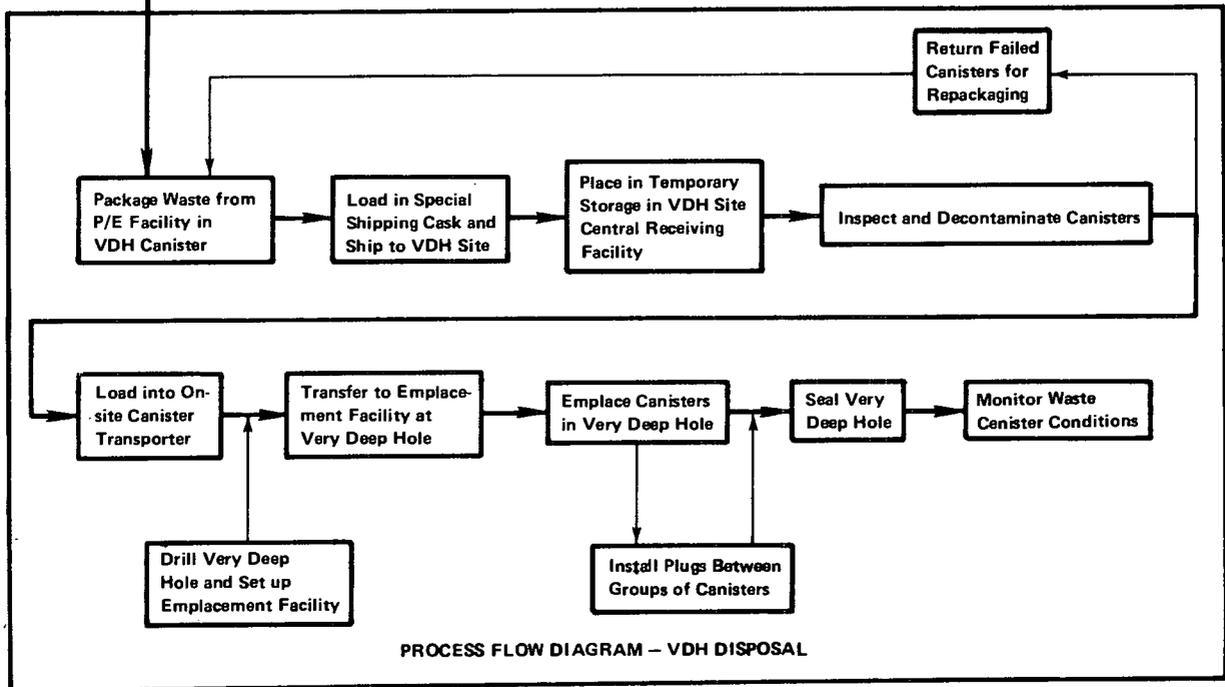


FIGURE 6.1.1. Major Options for Very Deep Hole Disposal of Nuclear Waste



FUEL CYCLE DIAGRAM - VERY DEEP HOLE DISPOSAL

HLW from Fuel Reprocessing Facility
or
Spent Fuel Assemblies



PROCESS FLOW DIAGRAM - VDH DISPOSAL

FIGURE 6.1.2. Waste Management System--VDH Disposal

6.1.1.2 System and Facility Description

System Options

The reference concept for the initial VDH disposal of nuclear waste has been developed from a number of options available at each step from the reactor to disposal in the very deep hole.

Various options to be considered for VDH disposal are summarized in Figure 6.1.1. The bases for selection of options for the reference concept (those blocked off) are reviewed in detail in various documents listed in Appendix M.

Because options for the waste disposal steps from the reactor up to, but not including, the geologic medium are similar for mined geologic repositories and VDH disposal, the options selected for the reference design are similar for the two concepts. From that point on, the options selected for the reference design are based on current program documentation for VDH disposal.

Waste-Type Compatibility

Very deep hole disposal would be limited to unprocessed spent fuel rods and the HLW from uranium-plutonium recycle cases. Because of cost constraints, VDH disposal of contact handled and remotely handled TRU wastes is not considered likely. Handling the large volume of these wastes would substantially increase drilling activities, costs, and the extent of adverse environmental impacts for VDH disposal. Thus, the low- and intermediate-level TRU wastes would require some other form of terrestrial disposal. It is assumed for the reference case that these wastes would be placed in mined geologic repositories.

Waste-System Description

The reference concept design was selected through judgment of a "most likely" approach based on available information and data. The fuel cycle and process flow for the reference concept are shown in Figure 6.1.2. In the reference concept, a VDH repository is designed for disposal of 10,200 canisters per year of spent fuel or for 2,380 canisters per year of solidified HLW. With a 40-year repository operation period, emplacement of spent fuel would require 68 holes per year with 150 canisters placed in each. Multiple holes would be drilled while others are being filled. HLW would require emplacement of 375 canisters per hole in six to seven holes per year (Bechtel 1979a), also with simultaneous drilling and emplacement operations.

Predisposal Treatment and Packaging. The predisposal treatment of waste for the VDH concept would be identical in many respects to the predisposal treatment of waste for the mined geologic repository concept. Chapter 4 of this document discusses the predisposal systems for both spent fuel and HLW common to all of the disposal concept alternatives.

The specific waste form required for emplacement in the deep hole is not yet identified. The waste form and canister would have to be structurally strong to resist downhole stresses and crushing forces, and chemically resistant to the waste emplacement medium. A metallic matrix or a granular waste form would be possible (Bechtel 1979a).

The canister would have to provide for safe handling, shipping, and emplacement of the waste. Both the HLW and the spent fuel canisters would have to be packed solidly to avoid crushing due to hydrostatic pressure of drilling "mud" (lubricant) left in the hole to counter lithostatic pressure. The canisters and spacers would have to be dense enough to sink through the mud slurry to the bottom of the hole. Carbon steel is considered as one candidate canister material that will fulfill these requirements (Bechtel 1979a). However, more complex designs using multiple barriers may be required.

The canisters for both HLW and spent fuel would have to be small enough for emplacement in a hole lined with a steel casing. HLW canister dimensions identified for the reference case accommodate the fuel. Dimensions identified for the reference case are 36 cm (14 in.) diameter and 4.6 m (16 ft) long (Bechtel 1979a and TID 1978).

Site. The critical geologic parameters that will determine the feasibility and impact of nuclear waste disposal in a deep hole system and that must be considered in site selection are:

- Lithology
- Tectonics and structural setting
- Hydrologic conditions
- States of stress
- Mechanical properties of the rocks at depth
- Natural thermal regime
- Geochemical reactions.

The interactions of these parameters and the effect of heating by the waste (thermomechanical factors) may also be significant. Geologic assumptions underlying the VDH concept are that the hole will be drilled, or a shaft excavated, in a regime of moderate to low geothermal gradient in rock with high strength and low permeability. Furthermore, the wastes are to be deposited irretrievably - not stored (LBL 1979). The specific geotechnical considerations are addressed in detail in LBL (1979) and Brace (1979).

Since more holes would be needed, emplacement of spent fuel during a 40-year period would require a total land area of approximately 140 km². Canisters would be shipped by rail from a processing and encapsulation facility to the repository site, which would consist of a number of drilled holes around a centrally located receiving facility (Bechtel 1979a).

Waste Receiving Facility. The central waste receiving facility at the deep hole site would be used to unload the waste canisters, store them temporarily, and perform any work required to assure prompt emplacement in the hole. The receiving building would contain a cask handling area, a canister storage area, a hot cell, and auxiliary facilities (see Bechtel 1979a).

The cask handling area would contain facilities for receiving, cleaning, and decontaminating shipping casks and for reloading empty casks on rail cars. Upon arrival, an overhead

bridge crane would remove the loaded shipping cask and move it to the confinement section of the building. The lid would be removed and the cask aligned with a hot cell port. The HLW or spent fuel canisters would be removed remotely to a storage rack within the hot cell.

An interim dry storage area adjacent to the hot cell would have space for a 1-month supply of canisters.

The hot cell would include space for checking the canisters for visible damage, radiation leakage, and surface temperature. Facilities would be provided to decontaminate waste handling equipment in case of a canister failure. Damaged canisters would be overpacked and returned to the processing and emplacement facility for repacking.

The receiving facility would also provide auxiliary services such as ventilation, equipment maintenance, and a control system.

Canister Transporters. Canister transporters, similar to those used for subsurface transportation and emplacement in the mined geologic repository (Section 5.4), would be used to transfer the waste from the receiving facility to the emplacement facilities. Each transporter would consist of a wheeled vehicle suitable for operation on site roadways, a shielded transfer cask, and equipment for raising and lowering canisters in and out of the transfer cask. In the receiving facility, the transporters would be positioned over a portion of the hot cell to bottom load the canisters into the transfer cask. At the emplacement facility, the transporters would be positioned over the temporary storage area and the canisters would be bottom discharged into temporary storage.

Drilling System. The drilling rigs would be similar to those used in the gas and petroleum industries and would be portable for movement from one hole location to another on the site. Each complete rig would require a clear, relatively flat area, approximately 120 x 120 m (400 x 400 ft), at each hole location (McClellan 1977).

In the reference concept, the drilled hole for spent fuel is 48 cm (19 in.) in diameter and 10,000 m deep (Bechtel 1979a). For HLW, the hole is 40 cm (16 in.) in diameter. The depth and diameter, however, could vary depending on the geologic medium, the depth required to satisfy containment requirements, and the drill rig capacity. For HLW, the hole would be fully cased to the required depth with seamless steel pipe about 40 cm in outside diameter, which would reduce the hole diameter available for waste.

Oil field rotary drilling techniques would be used to sink the holes, which may be stepped down in diameter as the depth increases. To seal the pipe to the rock, a grout would be forced through the pipe and then back up between the wall of the hole and the outside of the casing. The bottom of the hole would be sealed.

During the drilling and emplacement operation, the hole would be kept full of drilling mud to facilitate drilling, prevent casing and canister corrosion, minimize casings sticking to the sides of the hole during installation, and counter lithostatic pressure.

Emplacement Facilities. Each emplacement facility would include a confinement enclosure to provide a controlled environment for emplacement operations, and the temporary canister

storage facility (Bechtel 1979a). The entire emplacement facility would be on rails for movement from hole to hole on the site.

As described above, canisters would be transferred from the receiving facility to the temporary storage facility, which would provide shielding and an accumulation area for canisters to accommodate differences between transfer and emplacement operations. Emplacement equipment with cable totaling at least 10,000 m in length would lift a waste canister from temporary storage into a shielded cask, position it over the very deep hole, and lower it through the bottom of the cask into the hole (Bechtel 1979a). The waste canisters would be lowered into the lower 1,500 m (5,000 ft) of the hole with metallic honeycomb spacers placed between each canister to absorb impact in case a canister is dropped (Bechtel 1979a). If required by canister structural design limits, a structural plug, anchored to the sides of the hole, would be emplaced between groups of canisters to support the load.

Sealing Systems. After all waste canisters are in place, the hole would be sealed to isolate the waste from the biosphere. Sealing could include plugging both the hole and the damaged rock zones around the hole.

The components of the sealing system would have to have low permeability to limit nuclide migration and sufficient strength to maintain mechanical integrity for a specified period. Possible plugging materials include inorganic cements, clays, and rock. The specific material or materials would be selected for compatibility with the geologic medium and down-hole conditions (Bechtel 1979a). Plugging could be done with standard equipment typically used by a drilling rig crew. For final sealing and closure of the very deep hole, drill rigs, similar to those described for hole drilling, would be set up at the hole location.

Retrievability/Recoverability. Waste canisters would be retrievable as long as they are attached to a cable during the emplacement process. Once the canister is disengaged, it would become essentially irretrievable. Post-enclosure recovery is likewise considered nearly impossible.

6.1.1.3 Status of Technical Development and R&D Needs

Present State of Development

The status of equipment facility, and process development for different operational phases of VDH emplacement are considered below.

Drilling Techniques. Four methods to excavate a very deep hole have been considered. These are oil field rotary drilling, big hole drilling techniques, drill and blast shaft sinking, and blind hole shaft boring. The latter three methods are limited in the depths that can be attained at present and in the foreseeable future. They might have applications in specific geologic media but will not be considered further here since the possibility of their use appears remote for waste emplacement in this concept. For details on these concepts, see LBL (1979).

For oil field rotary drilling, standard oil field drill equipment would be used. In this method, a drill bit attached to a drill pipe is rotated from the surface, and drilling mud is circulated through the drill pipe to carry cuttings to the surface. The drilling mud also assists in providing borehole stability, provides lubrication and cooling, and minimizes pipe sticking. Substantial rotary drilling experience exists; however, most of the drilling has been in sedimentary formations.

At least the upper portions of deep rotary drilled holes would be cased; and, in fact, the entire hole may need to be cased for borehole stability, as in the reference concept (LBL 1979). As described there, cement grouts would be pumped from the bottom of the hole up around the steel casing to seal the casing against the drilled borehole. If the entire borehole were cased, then the hole could be bailed dry (depending on the depth of the hole), and could be left standing open for extended periods. If the bottom portion of the hole were not cased, it is unlikely that the borehole would stay open if the hole were bailed dry. Some fluid, probably with a density somewhat higher than that of fresh water, would therefore be required in the open hole at all times.

There is little experience at drilling in hard, crystalline rocks, although such rocks may pose no more, or fewer, problems than drilling ultra-deep wells in sedimentary rocks. A limited number of oil field rigs are capable of drilling to 8,000-m (25,000 ft) depths and beyond, and there are presently four rigs in the U.S. capable of drilling to a depth of 9,000 m. The bottom portions of such holes have been drilled with a 16.5 cm (6-1/2 in.) diameter bit, and the holes were cased to the bottom. There is some experience in drilling geothermal wells where formation temperatures are 30-0 C (approximately 600 F) as anticipated in VDH drilling; however, these wells have not been drilled much below 3,000 m (10,000 ft).

It is believed that deeper and larger diameter holes could be drilled. A maximum well depth of about 11,000 m (36,000 ft) in rocks where borehole stability is not a problem is believed possible, using a 20-cm (7-7/8 in.)-diameter bit for the bottom hole. Depths of 9,000 m could be achieved with 31-cm (12-1/4 in.)-diameter bits in crystalline rocks where no gas pressure exists. For very strong rocks, the bottom part of the hole might be left open. In fact, for the 31-cm-diameter hole, the bottom part of the hole may have to remain open because current rigs (with current casing) would not be able to set casing to the bottom of a 9,000 m hole. A drill rig with a 15,000-m (50,000-ft)-depth capability has been designed but not operated which would utilize the largest available components. It would provide a 22-cm (8-1/2-in.)-diameter hole at total depth (Drilling DCW 1979). Salt has been drilled successfully to about 4,600 m (15,000 ft); below this, borehole closure prohibits further drilling.

Emplacement. The technology for emplacing waste canisters is not fully developed at present. Some technology for emplacing items to depths less than 10,000 m exists. For example, the Deep Sea Drilling Project has a hydraulically operated down-hole device that disconnects the boring bits.

Sealing. Standard oil field practices for cementing in casing have satisfactorily isolated deep high-pressure gas zones from shallower formations and from the surface for time periods measured in decades. Plugs of cement or other materials have been emplaced in abandoned oil and gas wells, both cased and uncased, and have maintained integrity over similar periods of time. In these instances, it is not uncommon for the casing to corrode prior to plug deterioration.

Logging/Instrumentation. Borehole geophysical logging techniques in existence and currently under development will permit the logging and analyses of a number of parameters critical to the emplacement of radioactive waste in very deep holes. Caliper, acoustic, televiwer, and other borehole geophysical devices are regularly used to verify the presence and distribution of fractures in well bores. Electrical logs, neutron porosity loss recorders, and other devices are used to verify the presence of water. Temperature logs and spinner logs are used to detect water flow. While all of this equipment can be used from depths of hundreds to thousands of feet, none of these tools can function at the temperatures [between 200 and 300 C (390 and 570 F)] and pressures anticipated at depths around 10,000 m, because of the electronics contained in the probe.

While rudimentary development of in situ stress measurements has been accomplished, the down-hole techniques would require significant improvement.

Issues and R&D Requirements

Depth of Hole. The hole depth required for adequate isolation from the biosphere would have to be determined by the geologic medium of interest and by the history and physical condition of that medium. Sedimentary rocks in some instances are considered as potential VDH locations, but only where they are considered to be lower in elevation than circulating groundwater, such as deep basins or hydrologically stable synclines. Crystalline rocks may be the best geologic medium for VDH disposal. Usable hole depth in crystalline rock would be influenced by the depth of ground-water circulation within that rock. Ground-water circulation in weathered granite near the surface in a humid environment will generally be significantly greater than in fresh granite in an arid to semiarid environment.

R&D is required to determine the depth required in various geologic media to minimize the possibility of significant circulation to ground-water systems. The top of the emplaced waste would still have to be significantly below possible contact with circulating ground water, and would have to be properly plugged and sealed against such contact.

Drilling. The discussion of the present state of development of drilling makes it clear that emplacement of nuclear waste in very deep holes is not possible at this time given that (1) the waste canisters will be 31 to 36 cm (12 to 14 in.) in diameter and (2) the depth required for isolation from the biosphere may be as great as 10,000 m. If it is assumed that these two criteria are valid for the conceptual system, then a number of problems related to drilling would have to be solved to attain emplacement in very deep holes. The key issue is whether it will be possible to develop the technology to drill to 10,000 m with a bottom hole

diameter of approximately 48 cm (19 in.) so that a 36-cm canister could be placed in a mud-filled, fully cased hole.

No increase in the present capability to rotary drill deep wells is expected by the year 2000 without some very significant effort to develop new technology. Currently, there is no industry demand to produce the technology advancement necessary. If sufficient resources were available to advance technology, a 9,000-m hole with a 48-cm (19 in.) diameter might be attainable by the year 2000. Most of the hole would be cased; however, in high strength rocks without gas pressure, the bottom part of the hole might be left uncased. Technology improvements required to reach this depth include:

- New drilling muds capable of operating at temperatures of 370 to 430 C (700 to 800 F)
- High-temperature drill bits, either roller cone or diamond
- New drill pipe, including improved designs and use of improved (high-temperature) steels
- Improved support equipment, such as high-temperature logging and surveying tools and fishing tools
- Improved casing materials (high-temperature steels) and joint design
- High-temperature cements and surface pumps for pumping these cements.

Waste Form and Package Integrity. Criteria currently being proposed for waste forms and packages require total containment within the package for the time period dominated by fission product decay (up to 1000 years). The development of materials to retain their integrity for this period of time at temperatures that would be reached when the ambient rock temperature is 200 to 300 C and under geochemical conditions that would be encountered would require significant effort.

Heat Transfer (Thermomechanical and Thermochemical Factors). Under a normal geothermal gradient of 20 to 30 C/km (60 to 90 F/mi) ambient temperatures in excess of 200 to 300 C (390 to 570 F) are expected at a depth of 10,000 m. The heat released by radioactive decay of the emplaced waste would further increase the temperature of the surrounding rock. The magnitude of this induced temperature increase would be determined by the thermal properties of the rocks and the power output of the waste.

Because of the very large height-to-diameter ratio of the column of radioactive waste, the heat flux from the waste would be mainly in the radial direction, as from an infinite cylinder. The temperature within the heat source itself would be very nearly uniform and would drop very abruptly at the ends. Therefore, from a purely thermal point of view, this geometry would be very favorable. It takes 200,000 years for heat from 5,000-m depths to diffuse to the surface (DOE 1979). The thermally induced effects on the chemical stability and mechanical integrity of the geological formation and upward driving of the ground water would be the most critical issues.

The thermochemical behavior of rocks around a deep hole is not predictable at present. Since controlling factors would be the jointing, fracturing, and fluid content of the rocks,

thermomechanical behavior would need to be studied in situ. Heater tests in a variety of rocks at design depths would probably be necessary to understand the complex response to local high temperature of rock that is water saturated, stressed, and fractured.

Some aspects of thermomechanical behavior of rocks can be studied in the laboratory, however. Since fractured rock is in question, and since characterization of natural fractures is at present impossible, these laboratory studies would involve large samples of rock containing one or more joints, obtained by special sampling techniques. The samples may have to be large (dimensions of several meters). This would require extension of present laboratory testing techniques to test at conditions simulating the in situ environment. The areas where study would be particularly needed include:

- Thermal cracking and other forms of degradation of rock
- Thermoelastic response of intact and jointed rock over a long time frame
- Changes in permeability caused by heating a rock mass
- Two-phase transport of fluids in fractured rock
- Hydraulic fracturing in thermally stressed rock
- Thermal conductivity of hot, saturated thermally stressed rock
- Stress corrosion due to heated ground water in thermally stressed rock.

Emplacement. Most people engaged in drilling for resource exploitation feel that, to prevent collapse, the borehole would need to be kept full of drilling mud at all times. This would include the period during which the canister would be lowered for the waste disposal concept. Getting the waste canister to drop through the drilling mud could be difficult because of the close clearance between the casing and canister. The potential accidental contamination of the drilling mud and lowering cable should a waste package be ruptured would raise numerous questions regarding decontamination techniques and optimum loading methods.

Thus, in addition to a need for substantial research and development on improving the properties of the drilling mud, techniques and equipment would have to be developed to assure lowering and releasing the canisters at depths of 10,000 m and for decontaminating the drilling mud and cable in case of canister failure during this operation.

Isolation from the Biosphere. The principal issue of radioactive waste emplacement in very deep holes is the long-term isolation of the waste from the accessible biosphere (LBL 1979).

In addition to packaging, hole conditions, and hole sealing, a number of other conditions would have to be addressed before long-term isolation from the biosphere could be assured. Several of these involve geotechnical considerations, including:

- An improved understanding of the hydrologic regimes of deep crystalline and sedimentary rock units, including porosity, permeability, and water presence.

- An improved understanding of in situ rock mechanical properties under the high temperature and pressure conditions expected at the required depths and under unusual thermal loading conditions. These properties include strength, deformation, stress state, and permeability.

Additional R&D might be required in the areas of site selection, site evaluation, and geochemistry (LBL 1979).

Sealing. It is assumed that the sealing system for very deep holes must meet the same time requirement for sealing penetrations used by mined repositories. The primary purpose of the seal would be to inhibit water transport of radionuclides from the waste to shallow ground water or the surface for the specified time period. For integrity to be maintained, the sealing material would have to meet the following requirements.

- Chemical composition - the material must not deteriorate with time or temperature
- Strength and stress-strain properties - the seal must be compatible with the surrounding material, either rock or casing
- Volumetric behavior - volume changes with changes in temperature must be compatible with the enclosing medium.

The seal system would consist not only of plugs within the casing, but also of material to bridge the gap between the casing and surrounding rock. To minimize the possibility of a break in containment, rigorous quality assurance would be required during the placing of several high-quality seals at strategic locations within the borehole.

Therefore, research and development would be needed in two major areas - materials development and emplacement methodology - to ensure permanent isolation. Materials development would include investigating plugging materials, including special cements, as well as compatible casing materials and drilling fluids, which might be incorporated into the sealing system. Because the seal would include the host rock, these investigations should include matching plug materials with the possible rock types. It is conceivable that different plug materials would be required at different points in the same hole.

Emplacement methodology would have to be developed for the particular environment of each hole. Considerations should include all envisioned operations in the expected environment, casing and/or drilling, and fluid removal. Because the emplacement methodology would depend on the type of sealing material, initial studies of sealing material development should precede emplacement methodology development. However, the two investigations would be closely related and there should be close interaction between the two phases. In situ tests should be performed to evaluate plugging materials. Equipment developed should include quality control and quality assurance instrumentation.

Logging/Instrumentation. Proper development and operation of a VDH emplacement system would require the collection of reproducible, remotely sensed data on the geologic formation from the bottom of a borehole under high temperature and pressure. Existing logging tools are generally not designed to operate at temperatures exceeding 175 C (350 F).

Remote determinations of water content and flow and in situ stress would need to be addressed to permit preplacement assessment of down-hole conditions to facilitate VDH system design.

Much of the R&D work under way for logging and instrumentation equipment would be applicable to monitoring equipment for the waste disposal area (DOE 1979).

R&D Costs/Implementation Time

The total cost for research and development for this concept is estimated to be about \$730 million (FY 1978 dollars) as derived from DOE (1979). The major portion of this cost, or about \$600 million, would be for development of drilling techniques and equipment. The development activity described could be accomplished over a 12 to 15-year period.

Summary

Major uncertainties, shortcomings, and advantages of the concept are summarized below:

- The capability to drill with diameters up to 50 cm holes to a depth of 10,000 meters does not exist and would require a tremendous advance in the state of technology. However, should it be demonstrated that considerably lesser depths, e.g., 3,000 m, are consistent with the concept they can be currently achieved with holes of adequate size.
- The temperature, pressure, and chemical environment at depth would present a potentially very hostile environment for the waste package. Significant advances in materials technology might be required to ensure long lived package design.
- Corrective action, defined as retrievability of emplaced waste, would be unlikely after emplacement.
- The approach is probably not consistent with the philosophy of being able to demonstrate technical conservatism in that design margins are considered small.
- Current methodology does not permit adequate assessment of the at-depth emplacement environment, nor are criteria available for site selection.
- The extreme depth of the concept, and the resulting lengthy path to the biosphere might compensate for many of the drawbacks.

6.1.1.4 Impacts of Construction and Operation (Preplacement)

During the construction and operation phases, the environmental impacts of the VDH concept would be those common to other drilling and excavation activities. Drilling the hole would raise environmental considerations similar to those for drilling deep holes for oil and gas wells, for uranium exploration and production, and for geothermal and deep rock mining. VDH impacts for these phases would be: the conversion each year of several square kilometers from present land uses to drilling/mining and waste repository activities; disturbance and removal of vegetation; temporary impoundment of water in mucking and settling ponds; accumulation of tailings; alteration of the topography at, and adjacent to, the site; and socio-economic impacts on housing, schools, and other community services. No special environmental considerations beyond those required for normal drilling would be required.

Health Impacts

Radiological Effects to Man and Environment. As indicated earlier, two different waste forms could be considered for disposal in very deep holes: spent fuel in canisters and encapsulated processed high-level waste. A detailed description of these forms is contained in Bechtel (1979a). Additional assumptions are that both waste forms would have undergone a 10-year decay period prior to emplacement and that secondary TRU wastes would be disposed via a mined geologic repository.

The estimated total occupational whole-body dose from VDH disposal during routine operations would be 4,150 man-rem/yr for the spent fuel waste form and 6,260 man-rem/yr for the HLW form (Table 6.1.1). Of this, 910 man-rem/yr for the spent fuel waste and 920 man-rem/yr for the HLW form can be attributed to the emplacement of waste in the deep hole. The detailed breakdown of doses directly attributable to the VDH concept is presented in Table 6.1.2. Doses attributable to the naturally occurring radioactive materials released during excavation of very deep holes are not included in the estimates.

The estimate of the total nonoccupational whole-body dose from VDH disposal is 380 man-rem/yr for the spent fuel waste form and 180 man-rem/yr for the HLW form (see Table 6.1.1.). Only a very small portion would be contributed by the deep hole -- 7×10^{-6} man-rem/yr and 3×10^{-4} man-rem/yr, respectively, for the spent fuel and HLW forms.

Only nonoccupational doses have been estimated for abnormal conditions and these are presented in Table 6.1.3. Insufficient data are available to allow an estimate of the exposure to occupational personnel during abnormal conditions. It can be only assumed that the exposure would be within regulatory requirements. In this instance, the estimated total

TABLE 6.1.1. Radiological Impact - Routine Operation (Bechtel 1979a)

	<u>Whole Body Dose, man-rem/yr</u>	
	<u>Occupational</u>	<u>Nonoccupational</u>
<u>Spent Fuel</u>		
AFR	1580	320
Packaging and Encapsulation (P/E) Facility	1100	20
Transportation	80	40
Repository (secondary waste)	470	5×10^{-6}
Deep Hole	920	7×10^{-6}
Total	4150	380
<u>HLW</u>		
P/E Facility	4090	90
Transportation	210	90
Repository (secondary waste)	1030	2×10^{-5}
Deep Hole	930	3×10^{-4}
Total	6260	180

TABLE 6.1.2. VDH Concept - Occupational Doses During Normal Operation (Bechtel 1979a)

Operation	Whole Body Dose, man-rem/yr	
	Spent Fuel	HLW
Primary Waste Receiving	170	220
Damaged Canister Receiving/Processing	80	100
Surface Waste Management	40	70
Decommissioning	40	10
Primary Waste Placement	370	320
Interim Confirm. Building	30	30
Support/Overhead	180	170
Total	910	920

whole-body dose would not be applicable because the individual estimates given in Table 6.1.3 cannot be added algebraically. However, note that for both waste forms the potential for the highest exposure would be for a transportation accident, which is not an operation unique to the VDH concept.

Nonradiological Impacts. Nonradiological impacts should be comparable to those of any large construction project and those of industry during operation. Injuries, illnesses, and deaths common to such operations might be expected.

TABLE 6.1.3. Radiological Impact - Abnormal Conditions(a)

Operation	Whole-Body Dose, m rem/event
	(Nonoccupational)
<u>Spent Fuel</u>	
AFR	2×10^{-3} (b)
P/E Facility	3×10^{-1}
Transportation	1100(c)
Repository (secondary waste)	60(d)
Deep Hole	60
<u>HLW</u>	
P/E Facility	3×10^{-1}
Transportation	1100(c)
Repository (secondary waste)	60(d)
Deep Hole	70

- (a) Dose estimates imply consequences of a design basis accident. No probability analysis is included.
 (b) Design base accident (DBA) is tornado.
 (c) DBA is train wreck, in urban area followed by a fire.
 (d) DBA is hoist failure handling secondary waste.

The occupational hazards during normal operations of the waste disposal system would be expected to be no more, and maybe fewer, than the average associated with the various trade/professional workers required to operate the system.

In the case of routine operation nonoccupational hazards, the expected impact would not be detectable.

There are no specific data available to permit a quantitative estimate of the consequences of accidents that may arise. It is expected that abnormal occurrences such as fires, derailments, transportation accidents, and equipment failures common to industry would occur, but with reduced frequency. Consequently, the occupational impact would be expected to be less than that for industry in general.

Natural System Impacts

Currently available information is so limited that quantitative estimates of the radiological impact on the ecosystem are not available. However, it is expected that, during normal operations, the impact would be minimal, i.e., not greater than that for the mined geologic repository concept. Engineered safety features would be provided to ensure that the disposal system would operate in compliance with regulatory requirements. In addition, location of the waste in holes as deep as 10,000 m would increase the transport path to several kilometers more than that for the mined geologic repository. This would tend to further mitigate the consequences of radioactive waste leak, should it occur, by increasing the transport time.

Microfractures and other openings might develop in the vicinity of the hole because of the stress relief created by drilling or excavation. In addition, small openings might develop within the cement plug and between the plug and the hole wall if the bonding between the two were not adequate. Such channels would provide pathways for contaminated waters to migrate to the biosphere. If the hole were sited below circulating ground water, the primary driving force for migration would likely come from the thermal energy released by the radioactive waste. The travel time to the biosphere would therefore depend on the availability of water, the continuity and apertures of the existing and induced fractures, the time and magnitude of the energy released, geochemical reactions, and the volume and the geometry at the opening over which the energy persists. The lack of data on the presence of water and the properties of fractures in deep rock environments prevents making any estimate of the consequences to the ecosystem.

Nonradiological effects on the ecosystem might impact both water and air quality. Water quality might be affected by the discharge of treated wastewater to the surface water and by rainfall runoff from graded areas, rock piles, and paved areas. Air quality and meteorological changes would come from the generation of fugitive dust and the creation of reflecting surfaces. Air quality would also be affected by emissions from diesel-powered construction and transportation equipment, stack gases, and fugitive dust. The exact discharge quantities and runoff characteristics and the exact amount and type of construction equipment are not

available at this time. Parameters such as vehicle miles, surface areas of structures and pavement, soil characteristics, and size of stock piles are also unavailable. For each of these parameters, a qualitative estimate was developed where the water quality effects are based on total land requirement for the facility. The meteorology and air quality impact estimate was based on the number of construction sites, which represent a variety of dust and diesel emissions, and the number of operational emission sources (Bechtel 1979a). The estimates are given in Table 6.1.4.

Socioeconomic Effects

A complete assessment of the socioeconomic impacts of the VDH concept cannot be made at this time because few data are available. In addition, the data that are available can be used only inferentially. These data, which relate to operating employees and community facilities, indicate that impacts would be only moderate.

These inferences are based on a classification scheme where minor, moderate, and major correspond to less than 2,000 employees, between 2,000 and 4,000 employees, and more than 4,000 employees, respectively. For the community facilities two locations is minor, three to ten locations is moderate, and more than ten locations is a major impact.

Aesthetic Effects

As with socioeconomic effects, only minimal data are available for aesthetic effects and these data can be used only inferentially. The available data relate to visual effects only. In this case, the inference is that aesthetic impact would be moderate for both waste forms.

This inference is based on a classification scheme where:

- Minor = no permanent structures, facilities, or equipment more than 100 m high
- Moderate = one facility with permanent structures, features, or equipment more than 100 m high
- Major = more than one facility with permanent structures, facilities, or equipment more than 100 m high.

TABLE 6.1.4. Nonradiological Environmental Impact

Category	Spent Fuel	HLW
Water Quality Facility Area, ha	2400	800
Meteorology and Air Quality, number of construc- tion sites/operational sources	9/42	0/10

Resource Consumption

The consumption of major resources for each case has been estimated from available literature.

Energy. The estimates of energy consumption in the forms of propane, diesel fuel, gasoline, and electricity are presented in Table 6.1.5 for both the spent fuel waste form and HLW (Bechtel 1979a).

Critical Material Other Than Fuel. The estimated consumption of critical resources is presented in Table 6.1.6 (Bechtel 1979a).

Land. The estimated total land that would be required for a 5,000 MTHM/yr waste disposal system is 14,000 ha (35,000 acres) for the spent fuel waste form and 8,000 ha (20,000 acres) for the HLW form. In both cases, the estimated impact would be moderate.

International and Domestic Legal and Institutional Considerations

The international/domestic legal and institutional considerations associated with a VDH repository are expected to be of the same nature as those addressed for a mined geologic repository. (See section 3.3.2 and section 3.5.2)

6.1.1.5 Potential Impacts Over the Long Term (Postemplacement)

The potential for impacts over the long term would relate both to human activities and to natural phenomena. In turn, human activities could be related to the failure of engineered features or human encroachment. Natural phenomena, such as earthquakes and volcanoes, could also degrade the integrity of the waste repository. The heating, rock alteration, or thermo-mechanical pulsing that could be caused by wastes reaching critical mass are issues common to other geologic disposal alternatives. These aspects would be dependent on the specific rock and site characteristics, waste form, quantity, and spacing and could be evaluated only when these parameters have been defined.

Table 6.1.5. Estimated Energy Consumption

Fuel Type	Spent Fuel	HLW
Propane, m ³	2.3 x 10 ⁴	1.0 x 10 ⁷
Diesel, m ³	1.6 x 10 ⁷	3.4 x 10 ⁶
Gasoline, m ³	1.6 x 10 ⁵	1.2 x 10 ⁵
Electricity, kWh	2.0 x 10 ¹⁰	5.6 x 10 ¹⁰

TABLE 6.1.6. Estimated Consumption of Critical Resources

Material	Spent Fuel	HLW
Carbon Steel, MT	3.3×10^6	6.8×10^5
Stainless Steel, MT	8.4×10^4	2.3×10^4
Components		
Chromium, MT	1.4×10^4	4.6×10^3
Nickel, MT	7.5×10^3	2.0×10^3
Tungsten, MT	3.0×10^3	0.5×10^3
Copper, MT	1.3×10^3	1.9×10^3
Lead, MT	1.3×10^4	2.9×10^3
Zinc, MT	1.2×10^3	0.6×10^3
Aluminum, MT	1.3×10^3	1.2×10^3
Water, m ³	2.0×10^8	5.9×10^7
Concrete, m ³	1.9×10^6	1.3×10^6
Lumber, 10 ⁴ m ³	5.6×10^4	3.8×10^4
Clays, 10 ⁶ MT	9.2×10^6	1.5×10^6

Potential Events

The long-term impact of a VDH repository on the ground-water regime would be governed essentially by the nature of the deep ground-water system. Because of the great depth of emplacement and the larger volume of rock available to absorb the energy released by radioactive decay, the deep ground-water system probably would not be appreciably perturbed by the waste itself. If the deep hole were located within a recharge zone or in a zone of lateral movement, the distance to the biosphere along the path of flow might be so long and the velocities so low that isolation might be effectively achieved. Furthermore, the transport of radioactive contaminants by the flowing water would also be greatly retarded by the increased residence times and the increased time for interaction of the contaminant with the host rock.

Engineering Failure of Isolation Mechanism. The principal engineered isolation mechanism for this waste disposal system would be the containment seal. After emplacing the nuclear waste in the deep boreholes, the holes would be sealed to isolate the waste from the biosphere. This isolation would have to be sustained for tens to hundreds of thousands of years for HLW. Not only would it be necessary to seal the borehole itself, but consideration would have to be given to plugging any damage that could have occurred around the hole.

The loss of the integrity of this containment seal might provide a pathway for the waste into the biosphere. The impact on the environment resulting from such a failure could be

evaluated only on the basis of site-specific parameters. The lack of specific data prevents a quantitative evaluation. However, it is not expected that resulting impacts would be any greater than those for a mined geologic repository under comparable conditions and might be less due to the longer pathway of smaller diameter than a mine shaft.

Natural Phenomena. Another concern for the VDH concept in the long term would be the susceptibility of the ground-water system to tectonic changes and volcanic action. The very concept of the deep hole is aimed at minimizing such effects by increasing the distance to the biosphere as much as is technically feasible. Placement of the waste disposal site in a tectonically stable region would reduce the probability of such catastrophic events. Site-specific data would be required to quantitatively assess the impact of natural phenomena leading to degradation of the containment.

Inadvertent Human Encroachment. Human intrusions into the VDH repository in the long term could result from drilling, exploration, and excavations. Monitoring, surveillance, and security operations carried out after the repository were closed would provide an increment of safety against such occurrence. However, the physical depth of the VDH would in itself be expected to provide a significant deterrent against human encroachment.

Potential Impacts

The loss of integrity of the waste disposal system as a result of an engineered system failure, natural phenomena, or human encroachment might give rise to environmental consequences by introducing radioactive waste into the biosphere, which would result in radiological health effects. Similarly, ecosystem effects and nonradiological health effects are conceivable.

Radiological Health Effects. It is difficult to predict the nature of future events that would cause a breach of the barriers isolating the nuclear waste from the biosphere. Hence, it is assumed that the system would perform as designed for a prespecified period of thousands of years (Bechtel 1979a). After the period in which the isolation scheme performs as engineered, the barriers would be assumed to be susceptible to breach by:

- Normal degradation, due to expected, naturally evolving events, such as breach by an aquifer with the eventual leaching and migration of the waste
- Abnormal penetration, due to unexpected events, such as drilling or mining of the waste site by man.

The actual scenarios are described in detail in Bechtel (1979a). The radiological impact is expressed in terms of dose per year or dose per event in the case of the abnormal occurrence. The impacts are given in Table 6.1.7.

Ecosystem Effects. An evaluation of the effects on the ecosystem in the long term requires data that are presently unavailable. However, it is not expected that the impact on the ecosystem would be any greater than that for a mined geologic repository, and maybe less, since the radionuclides would be expected to take longer to reach the biosphere.

TABLE 6.1.7. Long-Term Radiological Impact of Primary Waste Barrier Breach

	Waste Type	
	Spent Fuel	HLW
Normal Events (mrem/yr)		
Whole Body	7×10^{-4}	7×10^{-4}
Bone	5×10^{-4}	5×10^{-4}
Abnormal Events (mrem/event)(a)		
Whole Body	Negligible	Negligible
Bone	Negligible	Negligible

(a) Dose is 50-year dose commitment from 1 year intake to the maximum exposed individual.

Nonradiological Health Effects. Although there are no specific data to evaluate the non-radiological health impact, it is expected that these impacts would be comparable to those found in the corresponding industries, e.g., mining, drilling, and excavating.

6.1.1.6 Cost Analysis

All cost estimates are in 1978 dollars based on January 1979 dollar estimates (Bechtel 1979a) less 10 percent.

The estimates are based on preliminary conceptual design data and were developed without the aid of previous cost estimates for this type of facility. Because of the high uncertainties in the cost of rotary drilled holes as large and deep as are called for in this VDH concept, the costs given should be considered only as preliminary estimates.

Capital Costs

On the basis of the waste system description, as presented in Section 6.1.1.2, the estimate of the capital cost for the spent fuel case is approximately \$2.3 billion. For the HLW case, a capital cost estimate is \$290 million (Bechtel 1979a).

Operating Costs

Operating cost estimates for the spent fuel case have been calculated per year for years 1 through 38 and then for phasedown years 39 and 40. These costs, which include VDH rotary drilling, moving emplacement structures, hole sealing, and receiving facilities operations, would be about \$1.7 billion for each year through the 38th year, \$1.6 billion for year 39, and \$0.8 billion for year 40.

For the HLW case for the same time periods, estimated costs would be \$210 million for each year through the 38th year, \$200 million for year 39, and \$260 million for year 40.

Decommissioning Costs

Total estimated decommissioning cost for the spent fuel case would be \$32 million. Total for the HLW case is estimated at \$11 million.

6.1.1.7 Safeguards

As noted, the waste types that can be handled in the VDH concept would be limited by volume constraints. Thus, choosing this alternative would require safeguarding two separate disposal flowpaths. The risk of diversion would be strictly a short-term concern, because once the waste had been successfully disposed of in accordance with design, the waste would be considered irretrievable. Physical protection of the sensitive facilities and transportation operations would be the most effective way to deny access to the waste for the short term, as is common to most waste disposal alternatives. For additional discussions of pre-disposal operations safeguards see Section 4.10.

6.1.2 Rock Melt

6.1.2.1 Concept Summary

The rock melt concept for radioactive waste disposal calls for the direct emplacement of reprocessed liquid or slurry HLW and remote-handled (RH) TRU into underground cavities. After the water has evaporated, the heat from radioactive decay would melt the surrounding rock, eventually dissolving the waste. In time, the waste-rock solution would refreeze, trapping the radioactive material in a relatively insoluble matrix deep underground. The waste and rock should achieve reasonable homogeneity before cooling, with resolidification completed after about 1,000 years. Rock melting should provide high-integrity containment for the radionuclides with half lives longer than this period. Spent fuel and secondary wastes (hulls, end fittings, and contact-handled (CH) TRU are not suitable for rock melt disposal unless they could be safely and economically put into a slurry for injection. Otherwise, they would be disposed of using some other form of terrestrial disposal, such as a mined geologic repository.

The waste-rock solidified conglomerate that would ultimately result is expected to be extremely leach resistant, to the extent that it might provide greater long-term containment for the waste isotopes than a mined geologic repository. Because less mining activity would be involved, the cost advantages could be substantial (Bechtel 1979a).

After emplacement, the waste would be considered to be irretrievable, although it could probably be recovered at great expense during the charging or waste addition period while cooling water was still being added. However, the recovery operation would become much more complex and expensive with time as the size of the charge increased (Bechtel 1979a).

There are several technological issues to be resolved and considerable R&D work would be needed before this concept could be implemented. Primary needs would be for better understanding of heat-transfer and phase-change phenomena in rock to establish the stability of the molten matrix and for development of engineering methods for emplacement.

6.1.2.2 System and Facility Description

System Options

The reference concept for rock melt disposal of nuclear waste has been developed from a number of options available at each step from the removal of spent fuel from the reactor to disposal in the rock melting repository.

Various options to be considered are summarized in Figure 6.1.3. The bases for selection of options for the reference concept (those blocked off) are discussed in detail in various documents listed in Appendix M. In addition, a number of options for variations within the concept were considered. These options could improve the concept by changing the cavity construction method or the waste form, or by eliminating cavity cooling (Bechtel 1979a and DOE 1979).

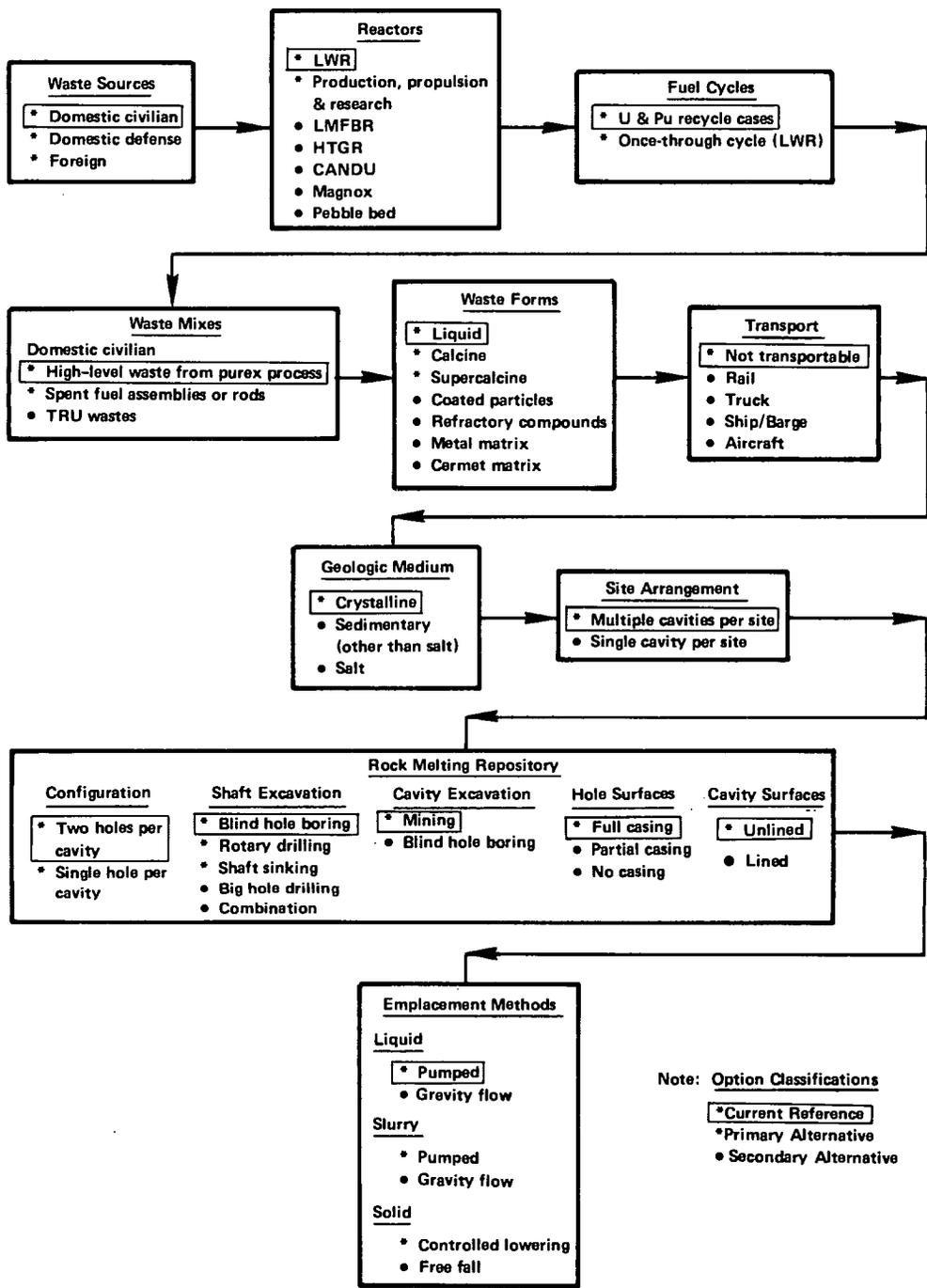


FIGURE 6.1.3. Major Options for Rock Melting Disposal of Nuclear Waste

Waste-Type Compatibility

It is assumed for the reference case that only liquid HLW and liquid RH-TRU would be injected into the rock melting cavity. Because of uncertainties associated with emplacement, such as additional criticality concerns, and a sufficient heat generation rate for the volume, spent fuel is not considered suitable for this reference case. Therefore, spent fuel and other wastes that may have low heat generation per unit of volume, such as solid RH-TRU and CH-TRU, are assumed to be sent to a geologic repository. Note that the suitability of spent fuel and other wastes for rock melt disposal may be improved by safely and economically putting them into a slurry form.

Waste-System Description

Basically, rock melting would work in the following manner. In the charging phase, HLW in aqueous solution would be injected into a mined cavity. The heat generated by the radioactive decay of the waste would drive off steam, which would be piped to the surface. When the boil-off rate reached a certain level, liquid transuranic wastes would be added to the charge. Periodically, high-pressure cleaning water would be flushed through the injection piping to minimize contamination and solid particle buildup. This cleaning water would also flow into the waste, providing a coolant to prevent the rock from melting during the waste charging phase. Cooling would be by evaporation or the heat of vaporization. At the surface, the steam driven off from the waste would be condensed and recirculated to cool the charge in the cavity. The closed system would be designed to prevent the release of radioactivity to the environment (Bechtel 1979a).

After about 25 years, when a substantial fraction of the cavity volume was filled, charging would be stopped. After the water was allowed to boil off and the waste to dry, the inlet hole would be sealed. The cavity temperature would rise rapidly and rock melting would begin, with radioactive materials dissolving in the molten rock. As the mass of molten rock grew, its surface area would expand and the rate of conductive heat loss to the surrounding rock would increase. Preliminary calculations indicate that at about 65 years, the rate of conductive heat loss from the melt pool would exceed the rate of heat input from radioactive decay. At this point, the melt would begin to slowly solidify. During the rock melting phase, the heat from the melt would inhibit ground water from entering the area and should prevent the leaching of the radionuclides. This is referred to as the "heat barrier" effect (DOE 1979). Following resolidification, when the heat barrier had dissipated, fission products would have decayed to very low levels. The relative toxicity of the residual radionuclides in the solidified waste-rock matrix is expected to be significantly less on a volumetric basis than that of a typical uranium ore from which nuclear fuel was originally extracted. The final product of the melt is expected to be a relatively insoluble sphere or resolidified silicate rock conglomerate, with a highly leach-resistant matrix, which would be deeply isolated from the biosphere (Bechtel 1979a).

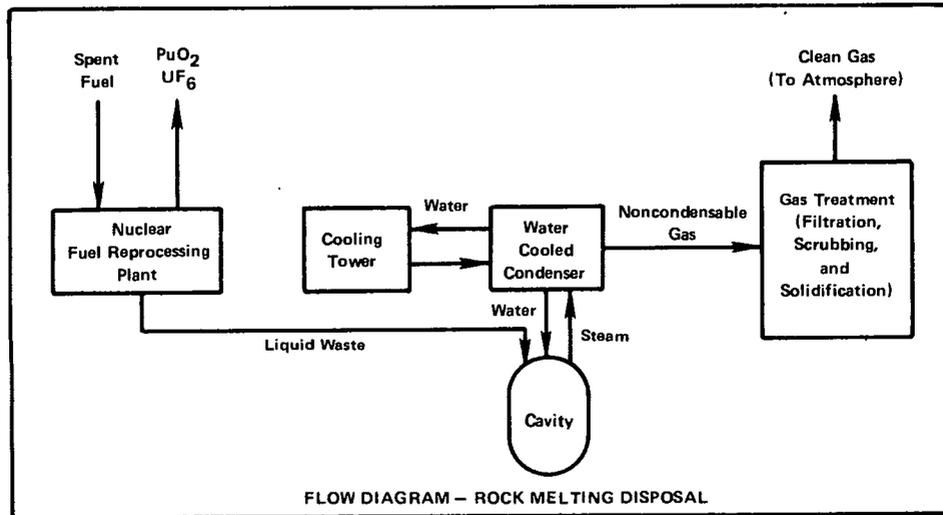
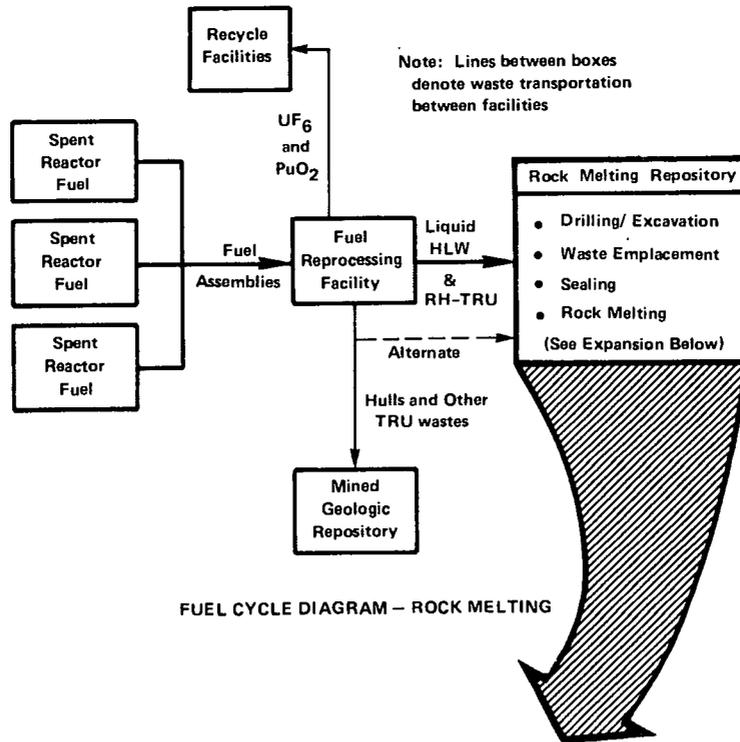


FIGURE 6.1.4. Waste Management System-Rock Melting Disposal

The reference concept design for rock melt disposal was selected through judgment of a "most likely" approach based on available information and data and is not supported by a detailed systems engineering analysis. The fuel cycle and process flow for this concept are shown in Figure 6.1.4. In the reference concept, a repository is designed for disposal of 4 million liters per yr (5,000 MTHM/yr) of high-level liquid waste (HLLW) for 25 years. This requires three 6,000 m³ (212,000 ft³) cavities, about 2,000 m (6,560 ft) below the surface on a single site. The three cavities would be located about 2,000 m from each other (Bechtel 1979a).

Predisposal Treatment of the Waste. The reference concept requires a fuel reprocessing plant to recover uranium and plutonium for recycle and to generate HLLW for disposal in the rock melting cavity, as described in Appendix VII of Bechtel (1979a). This plant could be located either on or off site, but the reference concept assumes an on-site location because of restrictions on the transportation of liquid radioactive materials. If solid pellets were produced in the packaging/encapsulation (P/E) facility, an off-site location would be feasible.

Site. The primary factor in selecting a site would be the suitability of the rock formations. Those rocks of greatest interest as potential media for rock melt disposal are composed of silicate minerals. Silicate mixtures are characterized by a melting interval rather than a definite melting point, the melting interval being different for each different set of minerals (DOE 1979).

The melting interval is bounded by the solidus temperature (the temperature at which liquid first forms as the rock is heated) and the liquidus temperature (the temperature above which mineral crystals do not exist stably). In rock melting, these temperatures would depend on parameters such as pressure, chemical composition (especially the amount of water present) and the state of segregation of the rock (see Figure 6.1.5) (Piwinski 1967, Luth et al. 1964, and Wyllie 1971a). Therefore, the ultimate size of the rock melt cavity would depend on the waste decay heat level and the rock characteristics, including thermal conductivity and thermal diffusivity. Also, the ultimate volume of the molten rock would be influenced by the size of the original mined cavity. The radius of the waste-rock melt pool, as a function of time, for a typical rock melt repository is shown in Figure 6.1.6 (DOE 1979).

The total site area that would be required for a rock melt repository would depend on the number of cavities, the size of the cavities, spacing between the cavities, and surface facility requirements. For this reference concept, the site area would be approximately 4 km² (1.5 mi²) (Bechtel 1979a).

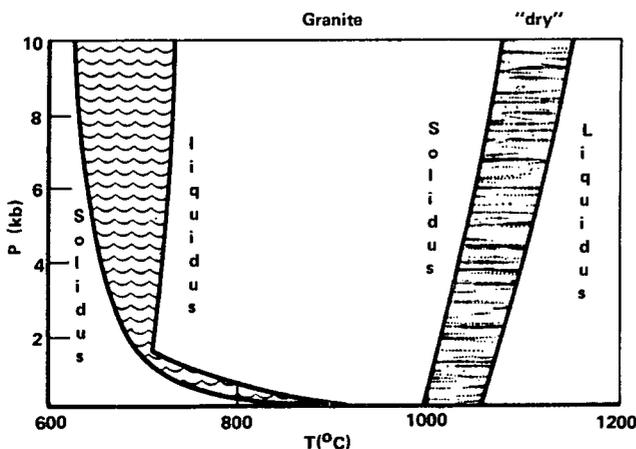


FIGURE 6.1.5. Schematic Illustration of Hydrous and Anhydrous Melting Intervals for an Average Granite

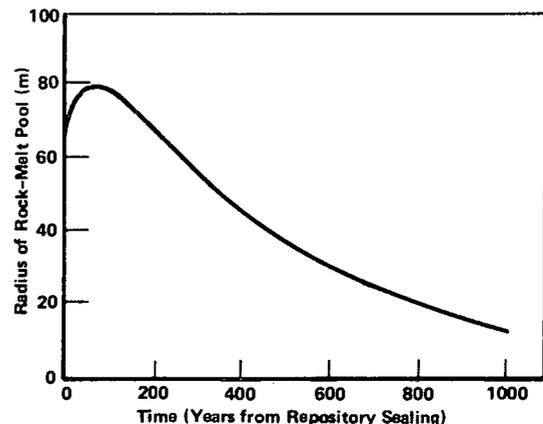


FIGURE 6.1.6. Radius of Waste-Rock Melt Pool Over Time (For Typical Cavity and Waste Loading)

Drilling/Mining System. The reference concept requires two access shafts for each cavity, each 2 m (6.6 ft) in diameter and approximately 2,000 m (6,560 ft) deep. They would be drilled using the blind hole boring method (Cohen et al. 1972). A rotating head with cutters would be turned by electric motors down hole. The entire boring machine would be held fixed in the hole by a hydraulic gripping arrangement. The shafts would be lined with carbon steel casings after drilling (Bechtel 1979a). This method would require men in the shaft to operate the boring machine (DOE 1979).

The cavity would be excavated by conventional mining techniques, although the equipment used would be limited by the access shaft diameter (Bechtel 1979a). Any blasting would be controlled to minimize fracturing of the surrounding rock. The spoil from both drilling and excavating would be hoisted up the access shafts by cable lift for surface disposal (Bechtel 1979a).

Repository Facilities. If the reprocessing plant were located on site, the reprocessing facilities would include a processing/packaging facility. If processing and packaging of wastes for off-site disposal were performed off site, the repository facilities would include a receiving facility similar to that described for the very deep hole concept (Section 6.1.1.1). The following description assumes that the reprocessing facility would be on site.

Four identical stainless steel tanks would be provided for storing HLLW. These tanks would have a combined capacity of about 10^6 liters (2.8×10^5 gal), which equals 3 months' production. The tanks, with the same design as those at the commercial reprocessing plant in Barnwell, South Carolina, would be contained in underground concrete vaults and provided with internal cooling coils and heat exchangers to prevent the waste from boiling (Bechtel 1979a).

An underground pipe system would connect the reprocessing facility to the storage tanks and the three rock melting cavities. The pipe would be double cased and protected by a concrete shielding tunnel. The pipe annulus would contain leak detectors. Heavy concrete and steel confinement buildings over the pipe and cavity shafts would provide for containment, shielding, monitoring, decontamination, maintenance, and decommissioning activities, primarily by remote control (Bechtel 1979a).

There would be four main pipes in the operating shaft to the rock melting cavity:

- A double-wall, stainless steel waste-addition pipe
- A single-wall, stainless steel water-cooling pipe
- A single-wall, stainless steel steam-return pipe
- A stainless steel instrumentation pipe through which monitoring devices would be inserted to measure the temperatures and pressures at various points in the system (Bechtel 1979a).

The confinement buildings over the cavities would also house the equipment and systems needed for filling the cavity and sealing the shaft. Three important process systems would

be: (1) the pipe and valve manifold enclosure, (2) the condensing plant, and (3) gas processing equipment. Pipe and valve manifolding would be located in an enclosure near the top of the cavity operating shaft. The cooling water injected into the cavity and the steam from the cavity would be routed through this enclosure. There would be an operating and instrumentation gallery adjacent to the enclosure (Bechtel 1979a). (The HLLW would be charged through a separate underground pipe, mentioned above, that would not go through the confinement building or the pipe and valve manifold enclosure.)

The condensing plant would cool and condense the steam coming out of the cavity and recycle it as cooling water during the waste charging phase. The potentially radioactive primary cooling loop and the nonradioactive, closed-circuit intermediate cooling loop, along with the associated pumps and heat exchangers, would be shop fabricated in modules and designed for rapid remote maintenance. Since the rock would start to melt in a matter of days without cooling, all heat exchanger and pump systems would be designed and constructed with full redundant capacity to ensure constant cooling.

Most of the gaseous elements in spent fuel would be removed during reprocessing at the fuel reprocessing facility. However, some fission product iodine in the liquid wastes could become volatile during the waste charging phase and would be carried out with the steam. This would be trapped by the gas processing equipment and returned with the cooling water to the waste charge or packaged for disposal in a mined geologic repository (Bechtel 1979a).

Auxiliary facilities would support the systems and equipment located inside the confinement building. These would include the water treatment plant, cooling tower, and radwaste treatment (Bechtel 1979a).

Sealing Systems. There would be two principal shaft sealing operations:

1. Sealing of the spare shaft after construction and before waste charging begins
2. Sealing of the charging shaft after completion of waste filling but before rock melting begins.

The NRC's Information Base for Waste Repository Design (NRC 1979) provides recommendations for sealing conventional boreholes and shafts. Though this information base may not be particularly applicable to the rock melt concept, it states that removal of the steel casing is essential for long-term performance of the seal. The seal must be bonded directly to the geological strata for maximum strength. Expansive concretes make the best seals under current technology and do so at an acceptable cost. However, it is not certain that these seals, whether cement, chemical, or other material, will successfully resist deterioration over a period of 1,000 years on the basis of current penetration sealing technology. Seal failure must be assumed even for seals placed under carefully controlled conditions using state-of-the-art technology and materials. Further development of sealing technology would, therefore, be required (DOE 1979).

Postplacement sealing of the pipes within the shaft, the shaft itself, and the pipes and valve gallery in the confinement building would be a more complex problem. This is be-

cause of the limited time, the high temperatures involved, and the radioactivity levels in the system. Considerable technology in this area has yet to be developed, as discussed in the following section.

Retrievability/Recoverability. Wastes disposed of by this concept would possibly be retrievable for a short period. Prior to melting, most of the liquid or slurry could be removed. After the melt has begun, well techniques for the molten rock-waste mixture might be possible. However this is unproven and would likely be an expensive and difficult process. Postclosure recovery of the solidified waste form would require extensive mining and excavation of large quantities of hot and molten rock containing waste.

6.1.2.3 Status of Technical Development and R&D Needs

Present State of Development

Substantial fundamental and applied research would be required for continued development of the rock melting disposal concept. This method is in the conceptual stage and no experimental work has been undertaken to support its feasibility.

Rock Melting Process. Generally, rocks are multiphase mixtures of a number of minerals characterized by a melting interval, as noted earlier. Because any two samples of a particular type of rock will have slightly different mineral compositions, they will also have slightly different melting intervals. As we have seen, the boundaries of these intervals (liquidus and solidus temperatures) depend on several parameters.

If the composition of the rock in which a waste repository were to be located has been well characterized, the melting properties of that rock could be predicted with some precision, and if the thermal conductivity, thermal diffusivity, and the heat of fusion of the rock were also known, the melting "history" of the HLW/rock melting phase could be predicted.

Clearly, it would be prudent to experimentally verify such predictions by means of prototype experiments; however, it should not be necessary to carry out an extensive series of such experiments to verify the current predictive capability for estimating the rate of rock melting and the total amount of rock melted for a particular set of waste repository conditions.

Effects of Heat on Rock Properties. The properties of rock subjected to high thermal gradients would be important inputs to determining the condition of the rock enclosing the molten waste-rock matrix. While the radius of this molten zone should be small compared with the extent of the geologic formation in which the repository would be sited, the zone's properties would have to be known so that an appropriate structural and safety analyses could be carried out.

The inner edge of this zone would be defined by the maximum radius of rock that had been heated to liquid formation. The outer radius of the zone could be roughly characterized as that location beyond which the rock had not been measurably affected by heat from the HLW.

The heat effects in the peripheral edges of the zone would be similar to effects found in a mined repository.

Transport of Radionuclides in Rock Melting. Under normal operating conditions, the casing in the emplacement well should prevent contact of radioactive waste with any aquifers that would overlie the disposal cavity. However, during waste charging, it is conceivable that some radioactivity could migrate out of the cavity into the surrounding rock. But, if the cavity were maintained approximately at atmospheric pressure, the tendency of water under hydrostatic pressure to flow into the cavity should minimize the importance of this transport mechanism.

During the rock melting phase, transport of radionuclides out of the waste-rock mixture would presumably be inhibited, because no water would be present in the melt and a portion of the surrounding zone of heated rock (Taylor 1977). (This is the "heat barrier" effect referred to earlier.) However, the radionuclide leaching capabilities of the high-pressure and high-temperature water vapor existing in this region would have to be characterized.

Finally, after the waste-rock matrix had cooled and solidified, it must be assumed that water would reenter the matrix and leach at least some of the radionuclides out of the matrix volume. Leaching potential at elevated pressure and temperature would have to be determined. As the radionuclides were transported to the relatively cool rock away from the repository, existing data on radionuclide transport in rock should be applicable (Klett 1974, Burkholder et al. 1977, de Marsily et al. 1977, Pines 1978, EPA 1978). It is possible that leaching data on other waste forms could also be useful (Brownell et al. 1974, Ralkova and Saidl 1967, Schneider 1971b, Mendel and McElroy 1972, Lynch 1975, and Bell 1971).

Effect of Superheated Water on Glasses in Rock Melting. Data from recent investigations of the devitrification of glass by water at high pressure and temperature (McCarthy et al. 1978 and McCarthy 1977) could be useful in determining the availability of radionuclides to water from vitrified rock present in the resolidified waste-rock matrix. However, the applicability of the conditions under which these data were obtained to the rock melt concept would have to be established.

Safety Studies: Disposal of HLW with Rock Melting. During the cavity charging portion of the presealing phase, HLW in such forms as solutions or slurries would be directly introduced into the repository cavity. The various operations that would be involved in carrying out this phase of the process are not as unique as the postsealing phase. Consequently, the probabilities for the release of radioactivity to the environment can be estimated for each step of this phase. This can be done both for normal operation and for assorted accident scenarios. In general, sufficient data exist to prepare a risk analysis for this phase of the rock melt concept.

After cooling of the waste-rock matrix to the point where water could contact the waste, it may be assumed for purposes of modeling that the waste dissolves, and transport through the surrounding rock is initiated. Calculations for risk analysis of this postsealing phase

are identical with those used for the risk analysis of other geologic waste disposal concepts with the exception of possible bulk migration of the molten mass during the interim phase between cavity sealing and solidification.

Ground Water Migration and Rock Melting. While a molten or high-temperature rock mass would disrupt natural patterns of water movement in the vicinity of a repository, the relative effect would diminish with distance, until, at some point, the repository would have no appreciable effect on water transport of radioactive materials. Presumably, if the hydrology of the repository area were well characterized, its effects could be modeled by treating it as a roughly spherical barrier with a radius that shrinks as the waste-rock matrix cools. Preliminary work on a laboratory scale and at atmospheric pressure indicates that this "thermal barrier" effect (Taylor 1977) could be demonstrated experimentally; however, additional work that more closely simulates conditions expected at the repository depth would be required.

Technological Issues

The technological issues that would require resolution before initiation of the rock melting concept can be summarized as follows:

- The necessary geological information cannot be predicted with present knowledge.
- Empirical data on the waste/rock interaction and characteristics are lacking.
- No technical or engineering work design of the required facilities has been attempted.

It is not possible at this time to produce a design for the rock melt repository because the necessary information is lacking. Data on the form and properties of the waste to be charged into the cavity, the charging methodology, the properties of the host rock, and many technical aspects of the shaft sinking method and cavity construction technique would have to be resolved. For many of these operations, work could not begin until fundamental waste/rock properties are better known.

In addition, the concept would require operations and process activities that do not readily lend themselves to the same degree of conservatism normally utilized in the nuclear field. Discussed below are several areas that would require further scientific or technical work.

Cavity Design and Construction. The greatest problem might lie in the construction of the cavity. Although, it is within the bounds of current technology to lower men and equipment through a 2-m-diameter shaft and construct the required cavity, such operations are difficult and time consuming. Methods for lining the cavity may have to be developed. Furthermore, it is practically impossible to construct the cavity without cracking the surrounding rock. Since it may be necessary to maintain the waste inside the cavity for some years before rock melting is permitted to begin, it would be necessary to ensure that waste does not escape into the cracks and ultimately into ground water. It may be difficult to assure

the necessary leaktightness of the mined out cavity. All of these areas would require technical resolution before construction could begin.

Cavity Charging. Cavity charging methods would depend on many variables including: the radioactivity of the charge; whether the charge were liquid or slurry; whether charging were batch or continuous; and whether charging were a long-term or short-term operation. The methodology for charging has not been defined or optimized. Considering the heat of the waste, the depth of the cavity, and possible corrosion and material plate-out, considerable technical effort would be required in this area.

In addition, the effect of a 2,000-m-long steam line on cavity charging would have to be determined. A vertical pipe of this length would act as a distillation column. Also, the engineering required to construct such a pipe (i.e., the number and type of expansion joints, effect of bends, etc.) has not been performed.

Shaft Sealing. There would be two phases of shaft sealing: sealing after construction but before waste charging starts and sealing after the waste is emplaced but before rock melting begins.

Sealing after construction would be the easier of the two operations because there would be sufficient time to check the work. However, sealing before rock melting begins would have to be done fairly quickly and in a potentially contaminated environment. Radioactive contamination and possible residual steam venting would present substantial problems in trying to seal the shaft after charging. Because of the number of pipes connecting the cavity to the surface, this operation would require considerable expertise. Both the materials and methods required would need further study and experimentation.

Volatile Fission Products. The quantities and behavior of the potentially volatile fission products would have to be determined. Nuclides in this category include ^{103}Ru and ^{106}Ru . Equipment would have to be designed to trap and remove these products from the waste stream or to return them in the coolant back to the cavity. Alternatively, they might be returned to the processing facility. There might also be a liquid and solid carryover from the steam, which would contaminate the condenser as well as increase the hazard from any potential leak. Practical technical considerations in this area would have to be examined before this concept could ever be considered viable. There is also a potential problem with tritium being carried with the steam.

Criticality Potential. Because 99.5 percent of the uranium and plutonium would have been separated from the spent fuel during reprocessing, the potential for criticality in the HLW is small. If experimental and modeling results indicated that criticality might be attained at some point in one of the rock melt concept scenarios, and if the results of such an excursion were undesirable from either an engineering or a safety standpoint, additional work would have to be carried out to develop methods of mitigation, possibly involving the addition of a high neutron cross section "poison" to the HLW as it is emplaced in the repository. It would be necessary for the "poison" to remain dispersed in the proper place upon cooling.

Fracturing During Cooling. During melting, the waste-rock mass would be expected to expand about 13 percent. During subsequent cooling and contraction, fracturing would have to be expected in the rock zone that surrounds the molten area. Further work would be required to establish that the rock melting concept could provide containment of the waste charge under uplift and subsidence conditions.

Chemical and Physical Effects on Surrounding Rock During Rock Melting. While the rock melting process can be described with some precision (Piwinski 1967, Luth et al. 1964, Wyllie 1971a, and Wyllie 1971b), the effect of a large thermal gradient on various types of rock has apparently not been similarly investigated (Executive Office of the President 1978). Although in some rocks, the predicted thermal effects of a molten mass of HLW/rock extend over relatively short distances, the extreme thermal gradient would clearly produce chemical and physical effects in the rock (Jenks 1977, National Academy of Sciences 1978). These effects would have to be characterized so that the rock mechanics of rock melt disposal could be adequately modeled and any possible intermediate or long-range effects identified and characterized. It would be necessary to carry out measurements over a range of pressures up to the maximum contemplated lithostatic pressure for a waste disposal cavity.

Interaction of HLW with Rock. At the present time, it is not clear whether the possible chemical reactions between the HLW solution and the rock cavity walls are important to the rock melt concept. However, it is clearly desirable to know how and to what extent such reactions take place, and to predict what the ultimate effect of 25 years of waste solution addition would be. With that information, potential problems could be identified, and mitigating measures could be designed and tested.

After addition of HLW to the cavity were stopped and rock melting begun, it is not known how rapidly and completely the HLW would mix with the molten rock. Because relatively complete mixing of the HLW with the rock appears desirable (to ensure complete dissolution of the HLW in the rock and subsequent immobilization upon resolidification of the matrix), it might be necessary to design the HLW rock melt disposal facility to minimize the viscosity of the molten rock.

Properties of Resolidified Waste-Rock Matrix. Even if it is assumed that the HLW is completely mixed with the molten rock, it is not known whether some of the radioactive species in the HLW might segregate during the long cooling process to form relatively concentrated (and possibly, relatively soluble) inclusions in the resolidified waste-rock matrix (Hess 1960). It is possible that the addition of certain chemicals (at the time that HLW is emplaced) could prevent such segregation, decrease the solubility of some or all of the long-lived radionuclides, or both.

R&D Requirements

Resolving these many uncertainties would require an extensive R&D program, such as that described below.

Data Base Development. Development of an adequate data base would require the conceptual design of one or more rock melt repositories. From these design bases, significant engineering features and critical geologic parameters could be identified. Similarly, the relevant properties of the geologic media would have to be understood in the context of the rock melt concept. Also, properties of materials in the waste handling systems would have to be identified and evaluated to determine the ability of these materials to function in hostile environments.

Laboratory-Scale Studies. To develop an understanding of rock melt mechanisms, extensive scale studies would need to be conducted. Specific areas of study should include:

- Heat transfer and phase-change phenomena for various geologic media
- Waste/rock interactions, particularly at elevated temperatures
- Properties of the resolidified waste-rock matrix
- Properties of engineering materials and their ability to function in the predicted environments
- Studies of actual small scale rock melt systems in laboratory hot cells
- Studies on the potential effects of criticality accidents.

Model Development. Better understanding of rock melt interactions could be gained by applying the data base to development of a predictive model covering heat transfer and related phenomena. The model could then be used for sensitivity analyses to determine the relative importance of various parameters and where research and development effort might best be applied.

Site Selection Methodology. From the systems modeling and other research tasks, it would be possible to identify those technological factors that would have to be considered in site selection. When site selection factors had been identified and evaluated, an optimal site profile could be determined to guide the selection process. Currently there is no methodology for locating a site.

Instrument Monitoring Techniques. Instrumentation for monitoring site selection and operational and postoperational phases of rock melt disposal would have to be identified and techniques for its use developed.

Thermal Analysis and Rock Mechanics. The effects of the melting cycle on the integrity of geologic formations would need to be thoroughly studied. Such effects as thermal expansion and contraction, phase change, and hydrologic change before and after emplacement would have to be assessed.

Pilot-Plant Studies. Laboratory and modeling studies should be complemented by a small-scale pilot-plant study involving actual emplacement of nuclear waste in rock. Such a study would be necessary to validate predictive methods and to assure that no vital factors had been overlooked prior to full-scale implementation of the concept.

Implementation Time and Estimated R&D Costs

In view of the significant technical uncertainties remaining, it is not possible to predict a cost estimate of the required R&D to implement this concept, nor the amount of time it would take.

Summary

Major uncertainties, shortcomings, and advantages of the concept are summarized below:

- There is not a multiplicity of engineered barriers inherent to the concept.
- The temperature, chemistry, and other characteristics of the molten waste-rock mixture are not considered consistent with technical conservatism.
- The required characteristics of a site are not known, and criteria for selection are considered extremely difficult to derive.
- The concept cannot be implemented in a step-wise, technically conservative manner due to the scale required for demonstration.
- Performance assessment capability is perhaps most distant for this concept than for any other.
- Retrievability of the waste is considered to be unlikely, so that corrective action cannot be accomplished.
- The time required for monitoring prior to full solidification (defined as the operational period of up to 1,000 years for this concept) exceeds the likely acceptable life for institutional controls.
- The primary postulated advantage relates to the possibility that the solidified waste form might be more stable than other possible forms.
- Lower mining requirements compared to a mined geologic repository may be a secondary advantage.

6.1.2.4 Impacts of Construction and Operation (Preemplacement)

Potential environmental impacts of a rock melt repository would be similar in many respects to those of a mined geologic repository. Both would require surface and subsurface activities that lead to environmental impacts. This impact analysis focuses on unique aspects of the rock melt concept, and refers to discussions on mined geologic emplacement in Section 5.4 as appropriate.

Health Impacts

Health studies related to the rock melt concept for the disposal of HLW can be divided into two phases: the presealing phase, which includes waste transportation and active operation of the waste disposal facility, and the postsealing phase, which includes the melting and resolidification of the HLW/ rock matrix and its long-term effects. In the following discussion, radiological and nonradiological concerns for the first phase are covered separately.

Radiological Impacts. During presealing operations, waste in solution or slurry form would be introduced directly into the repository cavity. Various operations in this charging phase could lead to release of radioactive material into the environment.

Under normal operating conditions, the casing in the emplacement well should prevent contact of radioactive waste with any aquifers that would overlie the disposal cavity. During waste charging, however, it would be possible that some radioactivity could migrate out of the cavity and into the surrounding rock. This possibility would be reduced if the cavity were maintained approximately at atmospheric pressure. Under these conditions, the tendency of water under hydrostatic pressure to flow into the cavity would minimize the importance of this transport mechanism. Nevertheless, it would be possible for radioactive material to reach man through such migration into the surrounding rock and onto the biosphere.

Operational impacts would vary somewhat, depending on which version of the rock melting concept is considered. If liquid HLW were emplaced directly into a cavity from the processing facility, there would be no impacts due to transportation of the waste. If solid waste were slurried into the repository, impacts of waste transportation from the reprocessing plant to the repository would have to be considered. However, such transportation would have no different environmental effects than would the shipping of such wastes to any other type of repository.

Treatment of HLLW prior to emplacement might be required to enhance the compatibility of the liquid with the rock in which the cavity would be located. This additional treatment step would increase the probability of occupational and population exposures to radiation. Handling and treatment of solidified HLW would also increase the probability of radiation exposure; risk analysis would take into account the details of the required handling and treatment procedures.

A summary of potential radiological health impacts was prepared for the rock melting concept (Bechtel 1979a). This study projected the short-term occupational impacts for a single rock melting cavity, which are presented in Table 6.1.8. For a 5,000 MTHM/yr throughput, it is estimated that three rock melting cavities would be required and that the impacts would be linear (Bechtel 1979a). Occupational impacts prior to the waste reaching the repository, nonoccupational impacts, and impacts from abnormal conditions were also postulated in this study. For this analysis, the consequence of impacts under abnormal conditions was found to be comparable to, or slightly less than, those of the other options. This study, however, did not include any probability analysis and consequently total radiological impacts under abnormal conditions have not been quantitatively determined.

Nonradiological Impacts. The underground portion of rock melt repositories would probably be constructed using conventional mining and drilling techniques. Health impacts would be those typical of any analogous construction project, and would be somewhat dependent on the method chosen (whether the cavity were created by mining, underreaming, explosive springing, etc.).

TABLE 6.1.8. Occupational Dose Estimate During Normal Operation
At a Single Rock Melting Cavity

Process Unit	Whole-Body Dose, man-rem/yr
Valve Gallery	120
Offgas Recovery	110
Maintenance	50
Decommissioning	30
Support/Overhead	40
Total	350

Impacts from surface construction would be typical of those associated with the construction of any chemical processing plant. Also, impacts similar to those for the mined geologic repository and discussed in Section 5.4 would be expected for this option.

Natural System Impacts

The effects of rock melting on ground-water migration and transport of radioactivity in the surrounding rock and the possible modeling of these effects are discussed in Section 6.1.2.3. This analysis suggests that heat from the wastes should not affect the thermal regime near the surface.

The principal impacts on natural systems associated with HLW disposal are considered to be those normally encountered in underground drilling and construction activities. Construction impacts could be estimated relative to those from conventional repositories on the basis of the amount of excavation required.

Such topics as disposal of mined spoil, emissions from machinery used in construction, and prevention of water pollution from mud pit overflow could best be analyzed for a specific site. General impacts, however, would be similar to those discussed in Section 5.4.

Because of the lack of formal studies, the effects of the melting cycle on the integrity of the geologic formation would need to be thoroughly studied. Effects such as thermal expansion and contraction, phase change, and hydrologic change during pre- and postemplacement environments would have to be assessed. These effects could be significant, but present data are insufficient to draw meaningful conclusions.

Socioeconomic Effects

Overall, the potential socioeconomic impact of a rock melt repository is rated as minor (Bechtel 1979a). This conclusion is reached, in part, because only a moderate sized work force (between 2,000 and 3,000 people) would be required for successful operation. Land requirements would be less than for any of the other disposal alternatives studied (Bechtel

1979a). In addition, with collocation of three rock melting cavities and three reprocessing facilities at each site, only two facility site locations would be required. The resultant fiscal impact on community facilities would therefore be relatively small.

Although rock melt might have the least socioeconomic impact of any of the alternatives, it is impossible to fully address the nature and extent of impacts at the generic level. This is particularly true when analyzing the socioeconomic impact of construction activity--a detailed estimate of the construction work force has not been completed. Nevertheless, it is reasonable to conclude that socioeconomic impacts would be similar to, and generally slightly less than, those described in Section 5.6 for the mined geologic repository. A cautioning note, however, is that collocation of facilities could lead to a concentration of impacts.

Aesthetic Effects

Facilities associated with a rock melt repository would have an aesthetic impact. The extent of this impact would depend on characteristics at the site and would reflect the fact that optimal engineering design would be necessary for different forms of HLW. Facility design would be a function of the physical and chemical form of the HLW.

The extent of surface construction would depend on the rock melting concept version for which the repository was being designed; where HLW solutions were being directly emplaced, the entire reprocessing plant would be located close to the repository. Where waste slurries were emplaced, only a relatively simple surface installation would be required to condense steam, add makeup water, provide for slurry mixing, etc. Aesthetic impacts would reflect final facility design, with larger facilities generally having greater impacts. Overall, aesthetic impacts would be similar to those described for a mined geologic repository, as presented in Section 5.6, with minor exceptions.

Facilities that would be different from those in the mined geologic repository include the type of cooling towers and tall drill rigs used in excavating the rock cavities. In addition, although a 100-m-high stack would be required for a processing facility, its location on the same site as the repository would reduce overall aesthetic impacts. Other aesthetic impacts, such as noise and odor, have not been identified as a problem with rock melt.

Resource Consumption

Energy would be required to construct and operate a rock melt disposal system. Initially, energy would be consumed in transportation and construction activities. In the operational phase, waste preparation, transportation, and emplacement activities would consume energy. Quantitative estimates of energy consumption for the construction and 40 year operation of a 5,000 MTHM/yr system have been prepared (Bechtel 1979a). These estimates are presented in Table 6.1.9.

Consumption of other critical materials has not been identified as an important factor in evaluating the merits of the rock melt concept. Drilling activities, as well as construction of the facilities, would require steel, cement, and other construction materials typically associated with a major facility. Estimates of these requirements are presented

TABLE 6.1.9. Estimated Energy Consumption (Bechtel 1979a)

Propane, m ³	1.0 x 10 ⁶
Diesel, m ³	1.5 x 10 ⁶
Gasoline, m ³	1.5 x 10 ⁵
Electricity, kWh	5.7 x 10 ¹⁰

in Table 6.1.10 (Bechtel 1979a). No scarce or otherwise critical material has been identified as being important for this option.

As noted, the reference concept calls for each rock melting repository site to support three 6,000 m³ cavities about 2,000 m below the surface (Bechtel 1979a). Each site would be able to accommodate waste from 5,000 MTHM/yr for 25 years. Construction of these facilities would disturb 1,100 hectares (2,720 acres) of land and would require a restricted land area of 4,000 hectares (9,880 acres) (Bechtel 1979a). Most of the land disturbed would be required for processing, encapsulation, and other surface facilities.

International and Domestic Legal and Institutional Considerations

The rock melting concept would have relatively few international implications because waste transportation activities would occur in the U.S. and emplacement would be achieved well out of range of the biosphere. There are, however, important domestic legal and institutional considerations that would need to be resolved. For example, as noted in Section 6.1.2.2, retrieval of wastes, even before emplacement activities were complete, would be very difficult. The hot nature of the wastes and the type of waste packaging that would be employed would influence the ease with which the waste material could be withdrawn. Retrieval after the cavity was sealed and the waste was in a molten form would be impossible. Legal and regulatory implications of these restrictions on retrieval would have to be resolved.

Selection of the rock melting concept would also affect certain decisions regarding interim storage. If waste from the uranium-only recycle, or the uranium and plutonium recycle were stored, it would be necessary to specify the form of waste storage that would have the least environmental and economic impact. Although it is possible that the waste

TABLE 6.1.10. Estimated Material Consumption (Metric Tons)

Carbon steel	300,000
Stainless steel	24,000
Components	
Chromium	4,800
Nickel	2,200
Tungsten	--
Copper	1,900
Lead	2,900
Zinc	600
Aluminum	900

would be stored as a liquid, it is more probable that it would be solidified (calcined or vitrified) if an extended storage period were envisaged.

6.1.2.5 Potential Impacts Over the Long Term (Postemplacement)

Although repository-related human activity would be minimal once emplacement and repository decommission activities were complete, impacts could occur because of the possible mobility of the molten waste material in the geologic environment. Potential events and impacts are described below.

Potential Events

For risk analysis purposes, the postemplacement phase of the concept is treated in a manner similar to other geologic disposal alternatives (see Section 5.6). As noted earlier, after the waste-rock matrix cooled to the point where liquid water could contact the waste, it is assumed that the waste would dissolve, and transport through the surrounding rock would be initiated. Clearly, the degree of risk calculated on this basis would be strongly site specific, and would depend on factors such as the depth of the repository, presence and location of aquifers, water quality, and sorptive properties of the rock.

Possible pretreatment of the wastes to minimize potential adverse postemplacement effects would depend on the waste form as well as the geologic media characteristics.

Potential Impacts

Basically, the environmental considerations involved in evaluating the long-term impact of rock melting are how much of the radioactivity in the repository would reach the biosphere, when it would get there, and what its effects would be.

The heat barrier effect is discussed in Section 6.1.2.3. Following total resolidification (1000 years), when the heat barrier no longer existed, most fission products would have decayed to innocuous levels. The toxicity of the residual radionuclides in the resolidified waste-rock matrix at that time should be significantly less than that of a typical uranium ore body from which the nuclear fuel was originally extracted.

Mixing of the HLW with the molten rock, as well as the physical and chemical properties of the cooled and resolidified waste-rock matrix, would determine the rate at which radioactive species could be leached and transported by ground water. It might be possible to design some mitigating measures to significantly retard leaching rates of all or some of the radioactive species present.

It is possible that the heat barrier effect would retard the start of effective leaching of radioactivity until radioactive decay had essentially eliminated the fission products as significant health hazards; thus, it might be necessary to consider only the TRU products.

Transportation of radioactivity by ground water would have to be evaluated on a site-specific basis, although different scenarios could be postulated to obtain order-of-magnitude estimates of the time required for radiation to appear in the biosphere and of the concentrations of radioactive species that would be present in the water. In modeling the

radioactivity transport, movement of water would be considered as taking place both through permeable rock and by means of joints and cracks in low-permeability rock (Heckman 1978). The impacts of a ground-water breach of a rock melt repository are expected to be similar to those that would result if a mined geologic repository were breached by ground water (Bechtel 1979a).

6.1.2.6 Cost Analysis

Cost estimates for the rock melt concept do not have the benefit of a reference conceptual design, nor of previous cost estimates for similar types of facilities. Therefore, these cost estimates are only approximate. They are based on the reference concept disposal of HLW from 5,000 MTHM/yr, for 25 years, requiring three cavities.

All cost estimates are in 1978 dollars based on January 1979 dollar estimates (Bechtel 1979a) less 10 percent.

Capital Costs

The capital cost of a rock melt repository with an operating lifetime of 25 years is estimated at \$560 million.

Operating Costs

An allowance of 2 percent of the capital cost is assumed for the annual operating cost, which comes to \$11 million a year.

Decommissioning Costs

The total decommissioning cost for the three-cavity rock melting concept is estimated at \$21 million. In this estimate, final shaft sealing is treated as a decommissioning cost with an allowance of \$2 million per cavity.

6.1.2.7 Safeguard Requirements

Because of the restrictions concerning the transportation of radioactive liquids, the fuel reprocessing plant would have to be colocated with the rock melt repository. Therefore, accessibility to sensitive materials would be extremely limited with liquid emplacement. If the waste were to be placed in a solid form (e.g., pellets), which could be emplaced in the subsurface cavity as a slurry, the fuel reprocessing plant could be located off site but transportation related safeguards would then be required. The subsurface cavity would increase the difficulty of diversion and the liquid or slurry waste form would complicate the transportation and handling problems for potential diversion. However unlikely, retrieval by drilling and pumping is possible. This would eventually need to be considered for rock melt repository safeguards. Material accountability would be enhanced by ease of sampling and measurement, but gross accountability (i.e., gallons vs. canisters) would be slightly more difficult than for the mined geologic repository concept. For additional discussion of predisposal operation safeguards see Section 4.10.

6.1.3 Island Disposal

6.1.3.1 Concept Summary

Island-based disposal would involve the emplacement of wastes within deep, stable, geological formations, much as in the conventional mined geologic disposal concept discussed in Chapter 5 with an over-water transportation route added. The island would provide port facilities, access terminals, and a remote repository location with possibly advantageous hydrogeological conditions. An island disposal facility could also provide an international repository if the necessary agreements could be obtained.

The island disposal concept has been referred to as an "alternate geologic approach" (Deutch 1978) in which the geology (i.e., rock, sediments) provides the primary barrier between the nuclear wastes and the biosphere and the ocean may provide an additional barrier, depending on the repository location and the hydrological system existing on the island.

The status of the concept is uncertain. The U. S. Department of Energy Task Force Draft Report (Deutch 1978) stated that "The Department of Energy has no program to actively investigate the concept. Suggestions for assessment of the concept have been made from time to time by groups considering international aspects of radioactive waste repositories. However, a consensus for the need of such repositories has not developed."

On the other hand, the sixth report of the U. K. Royal Commission on Environmental Pollution (Flowers 1976) referred to island locations when considering hard rock sites for a geologic facility. In this report, it was stated that "A deep disposal facility on a small uninhabited island would be particularly advantageous if one were chosen which was separated hydrogeologically from the mainland. Any leakage of radioactivity into the island's ground water would be easily detected and in that event the dilution of seawater would provide a further line of defense."

No detailed studies of the island concept are currently available; therefore, its basic elements are based on simplified modification and adaptations of conventional mined geologic disposal as discussed in Chapter 5. Since the geology of most islands is crystalline rock, it is the assumed disposal formation. Elements of other schemes (e.g., subseabed disposal, Section 6.1.4) have been incorporated and/or referenced where appropriate. If more detailed assessments are required in the future, conceptual design studies would have to be performed to provide a reliable basis for analysis.

6.1.3.2 System and Facility Description

System Options

The reference concept for the initial island disposal of nuclear waste has been developed from a number of options available at each step from the reactor to disposal in the island geology.

Various options to be considered for island disposal are summarized in Figure 6.1.7, with options for the reference concept designated. Details on the bases for selecting reference concept options are covered in various documents listed in Appendix M.

Because system options for island waste disposal beginning with the reactor and including steps up to the transportation requirements are similar to those for mined geologic repositories, the options selected for the reference design are similar for the two concepts. From that point on, the selected options are based on current program documentation.

Waste-Type Compatibility

An island repository could handle all wastes from the uranium and plutonium recycle case, and from the once-through cycle.

Waste-System Description

The reference island repository design is based on the concept discussed in Section 6.1.3.1 and the waste disposal cycle options identified above. The fuel cycle and process flow for the reference concept are shown in Figure 6.1.8. The reference system assumes the transport of all spent fuel, HLW and transuranic wastes to the island sites.

The waste forms and emplacement concept of canistered waste for island disposal would be the same as those for conventional mined geologic disposal discussed in Chapter 5.

Predisposal Treatment and Packaging. The predisposal treatment of waste for the island disposal concept would be identical in most respects to the predisposal treatment of waste for mined geologic repositories. Chapter 4 discusses the predisposal systems for both spent fuel and HLW common to all of the disposal concept alternatives.

Geologic Environments. The geohydrologic regime of an island, as diagrammed in Figure 6.1.9, comprises a self-contained freshwater flow system (called the freshwater lens because of its general shape), floating on a sea-fed, saline ground-water base. There are two possible locations for the repository--in the lens of freshwater circulation and in the deep, near-static saline ground water - shown as A and B in the figure.

Geographically, three classes of island have been identified:

- Continental Islands - located on the continental shelves and including igneous, metamorphic, and sedimentary rock types
- Oceanic Islands - located in ocean basins and primarily of basaltic rock of volcanic origin
- Island Arcs - located at margins of oceanic "plates", primarily of tectonic origin, and frequently active with andesitic lavas.

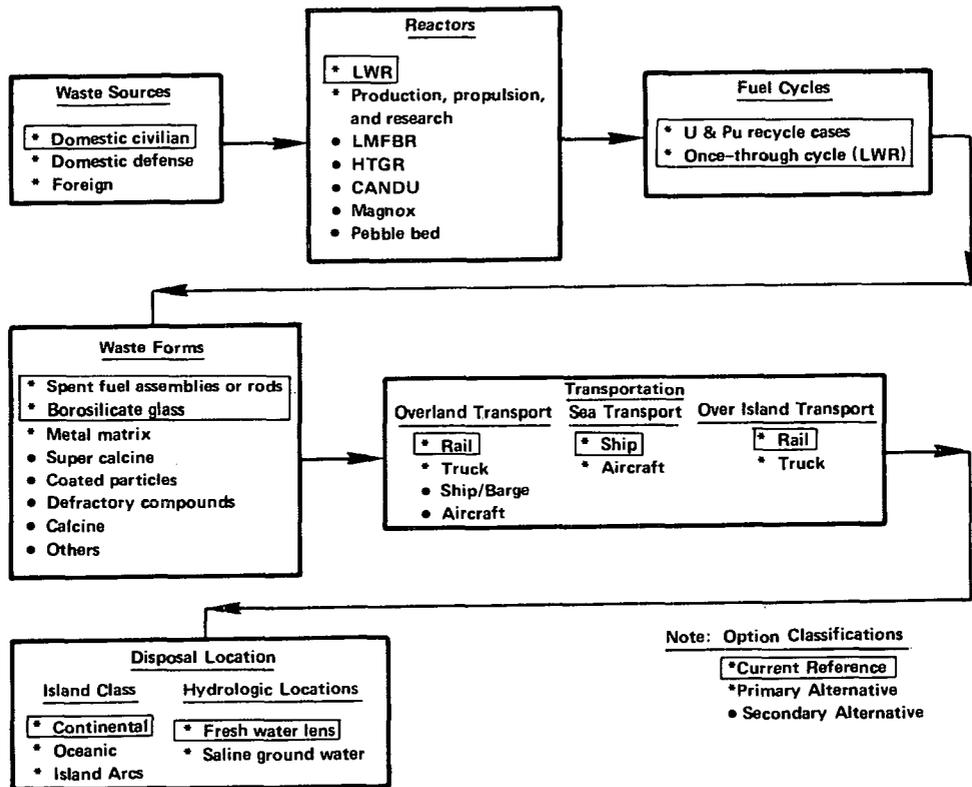
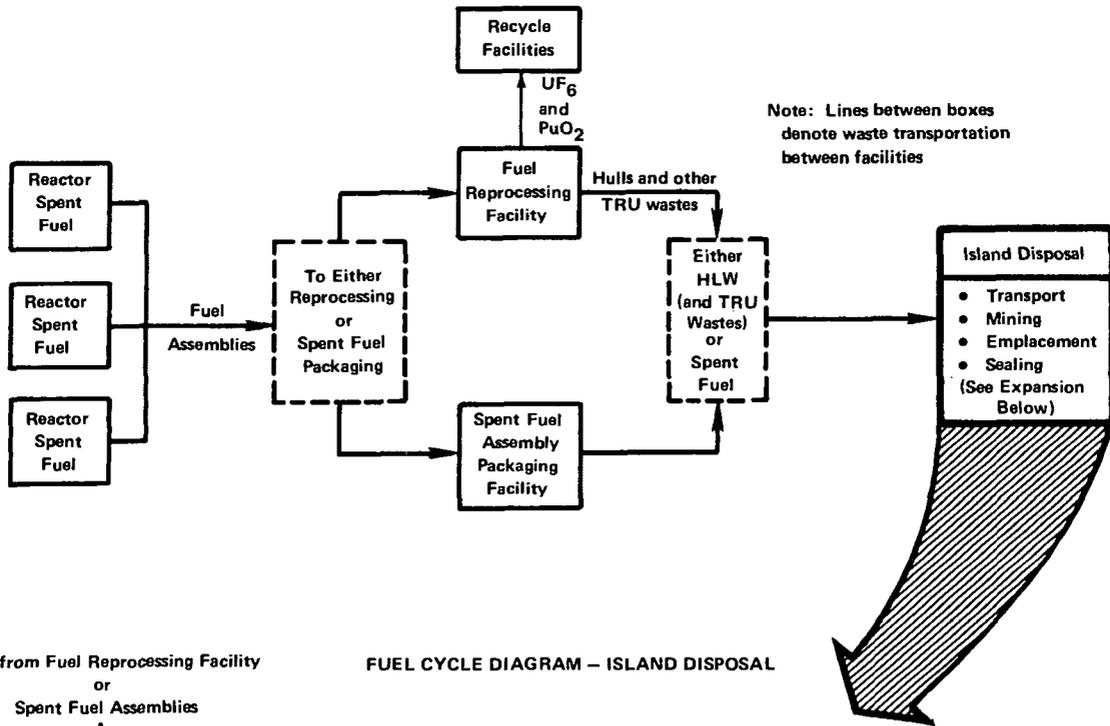


FIGURE 6.1.7. Major Options for Island Disposal of Nuclear Waste



HLW from Fuel Reprocessing Facility
or
Spent Fuel Assemblies

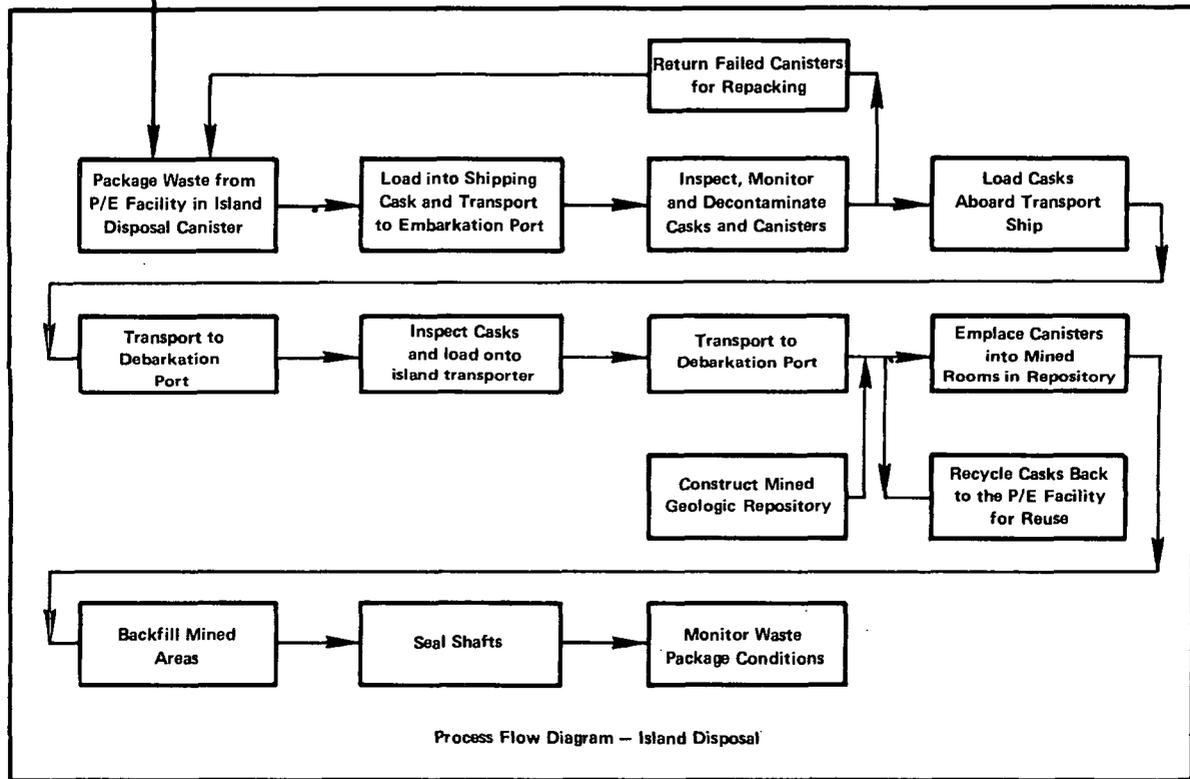


FIGURE 6.1.8. Waste Management System--Island Disposal

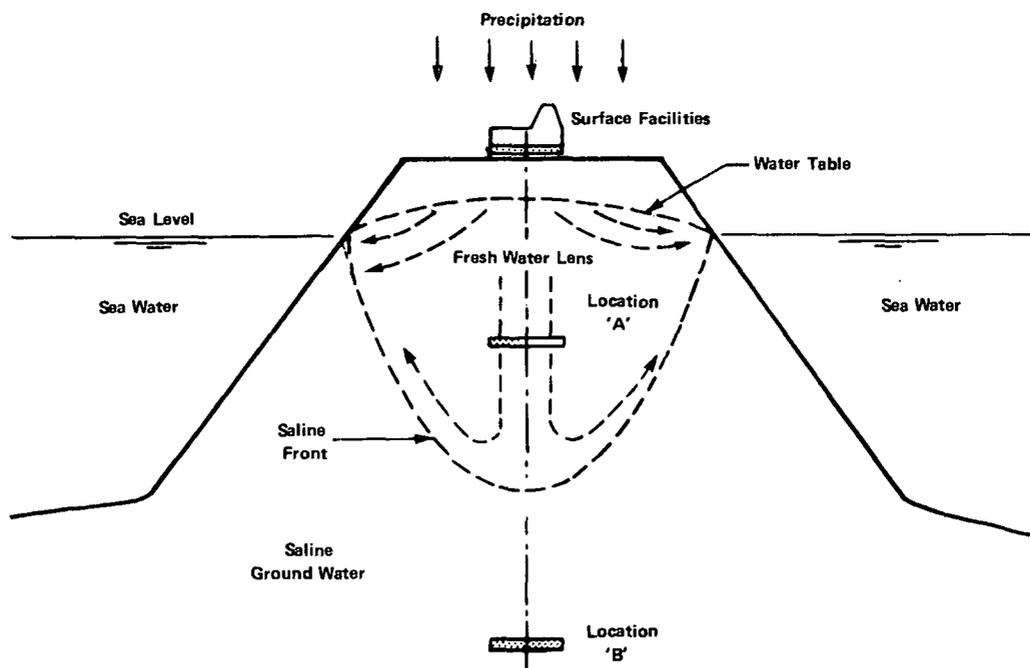


FIGURE 6.1.9. Hydrological Classification of Repository Locations

All three classes exhibit the classical island geohydrology described above, as modified by local geology and geographic setting. There are further discussions of the geology and hydrology of typical islands in DOE (1979), Todd (1959), Bott (1971), and Bayley and Muehlberger (1968).

Transportation Features. The island concept would incorporate the same basic procedure for transportation and handling as mined geological disposal. Of course, additional transportation from the mainland port to the island and additional receiving and handling facilities would be required. Transportation from the fuel reprocessing plant to the disposal site would be accomplished in three stages. The first stage would consist of truck or rail transport to a mainland port. Waste would be carried in transport casks that would cool the wastes and provide radiation shielding. (See Chapter 4 for a discussion of this procedure.) The second transport stage would be by ship to the island port. The subseabed disposal option (Section 6.1.4) details the operational features of this transportation phase. The casks would be cooled by either a closed-circulation water system, filtered forced-air system, or heat exchangers cooled by seawater. The coolant would be continuously monitored for radiation and temperature changes. Ship construction would provide for additional cooling. The ships could also include a shielded cell facility for examination of the casks.

The receiving port at the island would have the same features as the embarkation port described in Section 6.1.4. It could have a facility for temporary waste storage and transfer of the waste to specially designed transportation casks for final transport to the repository, the third phase. Conceptual design studies for island disposal are unavailable, but the required additional transportation facilities might be based on those discussed for the port and sea transport parts of the subseabed disposal option in Section 6.1.4.

Repository Facility. The layout of the reference repository for island disposal is a preliminary adaptation of the conventional geologic disposal concept discussed in Chapter 5. It is assumed that the island bedrock is crystalline and that the waste is emplaced approximately 500 m underground.

The conceptual design for an island crystalline rock repository is not supported by a data base comparable to that for salt repositories. The crystalline rock conceptual design discussed in Chapter 5 is assumed to be applicable to the underground aspects of island disposal except salt stockpile handling equipment would not be needed. The surface facilities for island disposal are assumed to be the same as for conventional mined geologic disposal.

Assuming that the repository capacity for spent fuel disposal is the same as for the conventional mined geologic disposal and that sufficient intermediate storage and transportation capacity can be provided, the once-through cycle would require four to eight island repositories, depending on the media. More repositories would be needed if island area were insufficient to support a repository of the size discussed in Chapter 5. Uranium-plutonium recycle wastes would require six to ten island repositories, depending on the island media (DOE 1979). The scheduled availability of the repositories for wastes from both fuel cycles would be expected to be a few years behind that of the conventional mined geologic disposal program.

Retrievability/Recoverability. Retrievability of emplaced waste or spent fuel from the rooms would be essentially the same as for the conventional mined geologic repository in crystalline rock. If retrieval were required because of deterioration or failure of the waste containers, special transportation containers and storage facilities would be needed. This need could be met by using a special cask design suitable for either rail, truck, or sea transport. Recoverability would also be similar to that with mined geologic disposal and would involve techniques similar to those used for the original emplacement process. Retrievability from island repositories could be complicated by the hydrogeologic characteristics of the sites.

Sealing, Decommissioning, and Monitoring. The sealing concepts might be the same as those for conventional mined geologic disposal in crystalline rock. The principal difference would be in the supply of labor and materials, which would involve sea transport to the island.

Final decommissioning of the island facilities could involve underground disposal of all contaminated equipment, the removal or disposal of all surface facilities, and suitable restoration and landscaping of the island.

Monitoring systems would be used during emplacement operations to detect air, surface water, and ground-water contamination. After the repository was sealed, a long-term monitoring system would be implemented. This system would be similar to those for the conventional geologic disposal concept, with modifications to suit the island option.

6.1.3.3 Status of Technical Development and R&D Needs

Present State of Development

In general, conventional mining techniques would be applicable to island repository construction. Transportation, storage, and handling requirements would be similar to those for the conventional mined geologic disposal concept, with the addition of the sea transportation link. Construction methods for ports would employ standard engineering practice.

Because the island disposal concept is so similar to the mined geologic repository option, the state of development is about the same. The ship loading and unloading requirements are similar to those described in the subseabed alternative, so again, the state of development is about the same.

Technical Issues

Technical issues that differ from those for mined geologic repositories lie in the areas of unique island hydrology and the resultant impacts of fresh or saline water on the package materials and the waste formulation.

For example: Is the waste form proposed for conventional mined geologic disposal appropriate for island disposal? Are the canisters that encapsulate HLW or the canisters of spent fuel compatible with the island repository environment? Should emplacement be in the freshwater zone or the saline ground-water zone?

Because a major incentive for considering island sites is a particular hydrological regime that frequently exists beneath them, efforts would be needed to:

- Verify the existence of a freshwater lens at various sites and determine its size.
- Determine the flow patterns and velocities of saline ground water at depths beneath the freshwater lens.
- Verify the stability of the freshwater lens in terms of the equilibrium between deep groundwater flows, salinity diffusion, precipitation and surface hydrology, the effects of sea level slopes, and other relevant processes in the natural state.
- Examine the perturbation to the lens caused by construction of the repository shafts and underground facilities, using simulation models and field evidence, if available. The shafts and facilities will tend to provide a sump that will drain either the freshwater or the saline ground water, depending on the location and depth of the repository.
- Examine the effects of heat generation on lens stability using simulation models. Heat may cause thermal convection cells that could flow counter to the freshwater circulation and modify the discharge pattern into the seawater.

R&D Requirements

To resolve these technical issues, specific R&D programs would be directed toward:

- Development of a system data base
- Study of hydrogeological aspects of island sites

- Development of criteria for and categorization of siting opportunities
- Risk assessment.

Implementation Time and R&D Costs

The time to complete the R&D, and the associated costs would be very similar to time and costs for a mined geologic repository. Increased R&D cost for the island concept would be expected to be a very small increment when compared to total costs for development of the mined geologic repository.

Summary

Major uncertainties, shortcomings, and advantages of the concept are summarized below:

- The transportation requirements to a remote location add to the overall risk of the concept.
- The state of knowledge relating to the hydrologic regime, upon which the concept relies, is not currently sufficient for siting or performance analysis.
- Considerable effort might be required to develop specialized waste forms and packages, if current reference concepts are not suitable.
- The approach does appear to be technically conservative if the hydrology is as predicted and to be capable of implementation in a step-wise manner.
- The concept employs the multi-barrier approach and has the additional attractive benefit of being remote.

6.1.3.4 Impacts of Construction and Operation (Preemplacement)

Impacts of construction and operation of predisposal systems in the island concept would be similar to those discussed in Section 5.6 for the mined geologic repository. Additional impacts from the sea transportation link and the port facilities would also be involved and are discussed in Section 6.1.4.4 for the subseabed disposal option. Impacts of mainland disposal are not discussed here.

Ideally, any island chosen for disposal would be totally uninhabited prior to construction of the repository (Selvaduray et al. 1979). In this case, the only non-occupational people impacted by construction and operation of the island repository would be families of those working at the facility.

Health Impacts

Radiological Impacts. Increased radiation exposure of occupational personnel under both normal and abnormal conditions would result from unloading of the waste at the receiving port, temporary storage of the waste, and transfer of the waste to the repository. Quantitative estimates of these exposures are not available at this time. However, unloading of the waste would probably result in exposures similar to those encountered during loading at the embarkation port, as discussed in Section 6.1.4.4 for the subseabed option. In addition, it is significant that the island repository would accept TRU wastes. This means that transportation impacts would be slightly greater than those for the subseabed option.

Moreover, although transportation-related impacts might be higher for island disposal, mainland benefits would be significant because of the elimination of the need to dispose of TRU wastes on the mainland.

The operation of the island repository itself is expected to be essentially the same as that for a mined geologic repository. Therefore, the exposure of occupational personnel to radiation should also be essentially the same. This exposure, during both normal and abnormal conditions, is discussed in Section 5.6.

In the event that there were any nonoccupational people on the island, the maximum dose received by any one of those individuals is expected to be similar to that received as a result of the operation of a mined geologic repository. However, because only a limited number of nonoccupational people should be present, total nonoccupational radiological health effects for an island repository are expected to be considerably less than those for a mined geologic repository.

Nonradiological Impacts. As indicated, impacts for island disposal should be similar to those of the subseabed and mined geologic disposal options. However, for an island repository in a relatively uninhabited area of the world, impacts would be significantly different from those of the mined geologic repository. In that case, potential non-occupational impacts would result primarily from transportation activities. Most transportation-related impacts are expected to be similar to those from the subseabed disposal option and are described in Section 6.1.4.4. That option, however, would not involve unloading waste material and increased transportation that could cause additional impacts from island disposal.

Natural System Impacts

Investigation of candidate island disposal sites would involve drilling and geophysical surveys, both on the island and in the adjoining offshore areas. During these activities, natural and wildlife habitats could be disturbed. Access and exploration operations could pollute both freshwater and seawater sources. Ecological effects could also arise from the use of explosives for seismic surveying. These impacts could be minimized by identification of sensitive areas and adequate planning.

Other ecological impacts, such as those described for the mined geologic repository in Section 4.8, would occur on the island selected for final disposal. However, because of the delicate balance of an island ecosystem, these impacts might require special consideration. In addition, the construction and operation of the required transportation and repository facilities would potentially impact the marine environment. These types of impacts have not been extensively evaluated.

Another important consideration is that small island ecosystems provide no refuge for the biota and ecosystems are much more easily affected by large-scale human activity. Furthermore, after the operational phase had ended, recolonization from outside sources would be far more difficult, and would take longer, than for a continental region. Finally, the types of

species that recolonize an island could be expected to establish considerably different trophic structures than were present prior to construction.

Emplacement operations in the repository would be similar to those for the conventional mined geologic disposal concept. However, if an accident were to occur within the island repository, water might be present because of drainage into the excavation. Thus, these operations, and other activities associated with the island repository, could affect the freshwater regimes on the island. In addition, water pumped from the underground excavation would be brackish if the repository were located below the freshwater lens in the saline zone. Therefore, care would be required to prevent contamination of surface freshwater streams and lakes. Disturbance of the natural ground-water regime could result in some freshwater wells becoming saline. Such activity could significantly affect the island's ecosystem, of which freshwater is a critical element.

Socioeconomic Impacts

Construction of an island repository would require assembling and transporting a large work force to a remote island. These activities would affect the socioeconomic structure of coastal communities through which the project personnel and equipment were transported. Detailed assessment of these impacts has been limited, but information presented on the subseabed and ice sheet options provides a useful perspective (Sections 6.1.4.5 and 6.1.5.5).

On the island, socioeconomic impacts would be a different type of concern associated with the entirely new communities that would normally be established. Selecting unoccupied islands for a final repository would greatly reduce socioeconomic impacts.

Aesthetic Impacts

Aesthetic impacts of the island disposal option would be limited because few people would live in the vicinity of the repository. During construction and operation, authorized site personnel would be the only individuals to perceive aesthetic impacts.

Aesthetic impacts would also be associated with transportation activities. Although these are generally not viewed as significant, additional discussion on this matter appears in Sections 6.1.4.5 and 6.1.5.5 on the subseabed and ice sheet disposal options, respectively.

Resource Consumption

Construction and operation of the island repository facilities would require energy, as would transporting the waste material to the disposal site, over mainland, ocean, and island routes. There are no studies available to quantify these energy needs.

Although the size of the facility and the land area required would be similar to that for the conventional mined geologic concept, it should be recognized that island repositories would likely require that an entire island be devoted to a waste repository. This commitment of land might not be important, however, considering that extensive study would be completed before an individual island was proposed as a disposal site.

International and Domestic Legal and Institutional Considerations

The island disposal option, like the subseabed and ice sheet options, would require transporting waste material over the ocean, and the general international implications of such transportation are important. Emphasis in this discussion is placed on aspects unique to island disposal.

Two, possibly complementary, international considerations would have to be studied for island disposal. On the other hand, an initial motivation for island disposal is that it could provide an international repository for use by many countries. On the other hand, the siting of a repository on an island over which the U.S. does not have sovereignty would require the approval of the nation that does.

International concerns could arise from countries in the vicinity of a proposed island repository. For example, if a remote island in the South Pacific were selected for an island repository, nations bordering the South Pacific might feel they were exposed to risks while receiving little or no benefit. Regardless of whether specific treaties were required, nations adjacent to any island disposal site would be likely to voice concern and seek international assurance of the safe operation of these facilities.

6.1.3.5 Potential Impacts Over Long Term (Postemplacement)

Potential Events

As in land disposal of radioactive waste, island disposal would require careful assessment of the processes by which the radionuclides could migrate from the containers through the various barriers to man's environment. Actual island emplacement of any quantity of such waste could occur only after the completion of a program to demonstrate, by analysis and experiment, the retention capabilities of each of the natural and man-made barriers to migration.

Waste Encapsulation. The waste form and canisters used for island disposal might be similar to those used in a mined geologic repository on the mainland. Studies of the specific effects of ground-water chemistry in either the freshwater lens or deep saline zones would provide data for establishing leach rates in the crystalline rock site.

Ground-Water Transport, Freshwater Lens Location. Waste emplaced in the freshwater lens might be exposed to the very slow ground-water circulation within the lens. The velocities would depend on rock permeabilities, porosities, precipitation, and surface hydrology. A simplified conceptual view of the potential pathways and barriers is shown in Figure 6.1.10.

Waste in the freshwater lens circulating system might be expected to discharge at the shoreline. Natural ground-water flow patterns might be affected by thermal convection and repository construction. Concentrations at the exit zone have not been estimated.

Radionuclides might be sorbed by the host rock, which would substantially retard the waste transport within the lens. Sediments that might exist at the shoreline in the discharge zone could have useful sorption properties and retard radionuclides prior to discharge and dilution in the seawater.

Ground-water Transport, Saline Zone Location. It has been suggested that offshore islands may have essentially static saline ground water at depth, due to the absence of hydraulic gradients at sea level. However, the residual or continuing effects of oceanographic, geothermal, climatological, or other changes may create flow. These effects would need to be examined prior to siting a repository in such a location (see Figure 6.1.11).

Flow transport in the saline zone may be accompanied by dispersion and diffusion, which would result in reduced concentrations at a distance from the repository. The amount of sorption of radionuclides in the host rock or on seabed sediments would depend on the particular radionuclide, ground-water, and rock or sediment chemistry.

Seawater Contamination. It appears that the principal discharge of wastes from an island repository would be into the seawater, possibly through sediments. Discharge might occur in a relatively concentrated near-surface zone if the waste were located in the freshwater lens. This could cause contamination of littoral and near-surface aquatic systems.

Discharge from wastes located in the saline ground-water zone would likely be dispersed through the seabed if the thermal-convection effects were insufficient to distort the flow patterns significantly.

Volcanism. Some islands, particularly those in island arcs and to a lesser extent oceanic islands, are frequently highly active seismically and volcanically. Such activity could discharge the waste in either lava flows or into the atmosphere. Geologic data for the most recent volcanic event would be relied upon to establish inactivity before an island was selected as a disposal site.

Potential Impacts

In determining the potential impacts of island disposal over the long term, the following factors would be considered:

- Corrosion, leaching, and transportation of radionuclides to the biosphere by the ground water
- The influence of thermal effects on flow
- Thermal/mechanical effects on permeability and porosity
- Retardation of radionuclides on rock fractures and seabed sediments
- Sediment and current movements
- Pathways to man via marine organisms, typical marine activities, and island considerations.

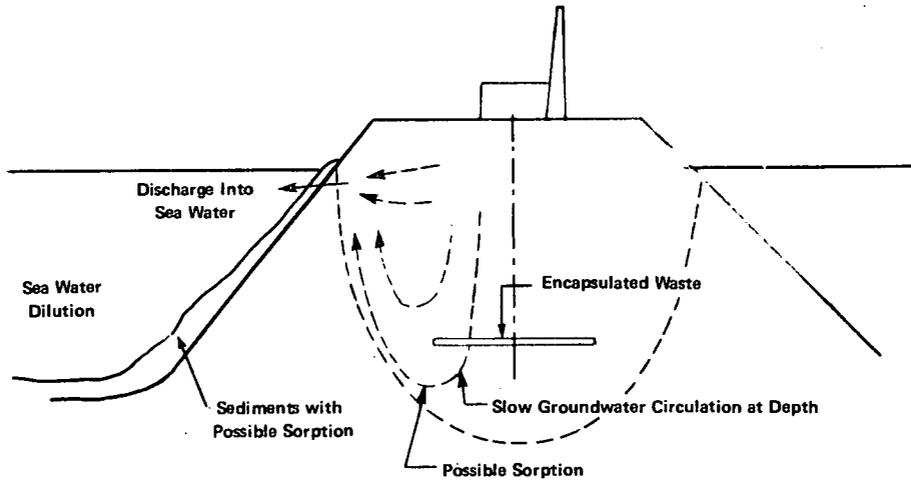


FIGURE 6.1.10. Isolation Barriers for Freshwater Lens Location

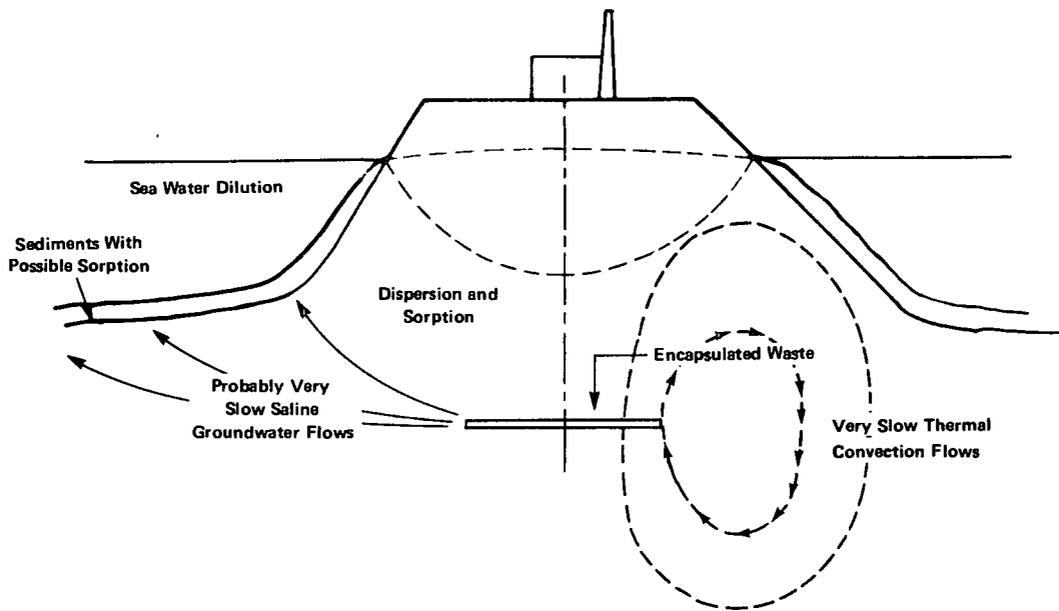


FIGURE 6.1.11. Isolation Barriers for Saline Zone Locations

Quantitative estimates of these impacts for the island disposal concept are unavailable at this time. However, it is expected that they would be similar to, but probably less significant than, those from a mined geologic repository. The reasons for the probable lessened impact are that (1) seabed sediments might provide significant sorption of certain radionuclides, (2) the sea would provide substantial dilution of discharges from the ground water, and (3) the island population, which would bear the greatest impacts, would be expected to be small in the long term because of the remoteness, size, and limited potential for inhabitation of any island that would be selected.

6.1.3.6 Cost Analysis

Detailed costs for island repository construction, operation, and decommissioning have not been estimated. It is estimated, however, that the cost of an island repository would be at least double that for a continental mined geologic repository because of sea transportation, the associated loading and unloading facilities, and the high salaries necessary for remote locations.

6.1.3.7 Safeguard Requirements

With the exception of ocean transportation, safeguard requirements for this concept would be expected to be similar to those for the mined geologic repository concept. However, the risk of diversion for the island disposal concept is primarily a short-term concern because of the remoteness of the disposal site and the major operational and equipment requirements for retrieval. Physical protection of the sensitive facilities and transportation operations would be the most effective way to deny access to the waste for the short term. For additional discussion of predisposal operations safeguards see Section 4.10.

6.1.4 Subseabed

6.1.4.1 Concept Summary

In subseabed disposal, wastes would be emplaced in sedimentary deposits of the ocean bottom that have been stable for millions of years. These deposits have a high sorptive capacity for the waste species (except for iodine and technetium) that might leach from the waste packages. Transport from ocean depths for any waste species escaping the sediments to the biologically active near-surface waters is expected to be a slow process that would result in dilution and dispersion. In addition, the great depth of the water column would constitute a barrier to human intrusion.

A program has been under way since 1973 to assess the technical and environmental feasibility of this concept for disposing of high-level nuclear wastes (Bishop 1974-75, Talbert 1975-78). The total seabed represents about 70 percent of the surface of the planet (of which less than 0.0001 percent would be used) and contains a wide variety of geologic formations. Theoretically, all wastes from the once-through cycle and uranium-plutonium recycle options could be emplaced in subseabed formations. But, because of volume considerations, other methods of disposal may be more practicable for contact handled and remotely handled TRU wastes.

The reference subseabed geologic disposal system for study purposes is the emplacement of appropriately treated waste or spent reactor fuel in a specially designed container into the red clay sediments away from the edges of a North Pacific tectonic plate, under the hub of a surface circular water mass called a gyre (mid-plate/gyre:MPG). (However, selection of the North Pacific as a study area in no way implies its selection as a candidate subseabed disposal site.) The reference method uses a penetrometer^(a) for emplacing wastes in the sediments in a controlled manner that allows subsequent monitoring. A specially designed surface ship would transport waste from a port facility to the disposal site and emplace the waste containers in the sediment. A monitoring ship, which would completely survey the disposal site before operations began, could determine the locations of individual disposal containers and monitor their behavior for appropriate lengths of time. The ship would also maintain an ongoing survey of the surrounding environment.

(a) A penetrometer is a needle-shaped projectile that, when dropped from a height, penetrates a target material. It can carry a payload of nuclear waste and instruments designed to measure and transmit its final position and orientation relative to the sediment surface. Penetration depth is controlled by the shape and weight of the penetrometer, its momentum at contact with the sediment, and the mechanical properties of the sediment.

6.1.4.2 System and Facility Description

System Options

The reference concept for the initial subseabed disposal of nuclear waste has been developed from a number of options available at each step from the reactor to disposal in the subseabed repository.

Various options to be considered for the subseabed concept are summarized in Figure 6.1.12. The bases for selection of options for the reference concept are detailed in sources cited in Appendix M.

Waste-Type Compatibility

It is assumed for the reference case that subseabed disposal is limited to disposing of spent fuel, HLW and cladding hulls. Other wastes are assumed to be disposed of in a mined geologic repository. However, it should be noted that these wastes may also be appropriate for subseabed disposal if there are sufficient economic incentives.

Waste-System Description

The reference concept design was selected as a feasible approach based on available information and data and is not supported by a detailed system engineering or cost analysis. The waste-management system, including the fuel cycle and process flow, for the reference concept is shown in Figure 6.1.13.

Subseabed disposal has as its foundation a set of multiple barriers, both natural and man-made, that would be employed to ensure the safe isolation of nuclear waste. These barriers are (Bechtel 1979a):

- The waste form
- The waste canister
- The emplacement medium (i.e., sediment)
- The benthic boundary layer
- The water column.

The water column is a barrier primarily to intrusion by man, although it would provide dilution and dispersion for radioactive species.

The waste form (leach-resistant solid) and the metallic waste canister or overpack would be man-made barriers. It is assumed that they could be engineered as a multibarrier system to contain the waste for a period during which the heat-generation rate due to fission product decay would decrease to low levels.

The emplacement medium (clay sediment) shows evidence that it could provide long-term containment of the nuclides through its sorptive qualities, ion-exchange characteristics, and very low permeability.

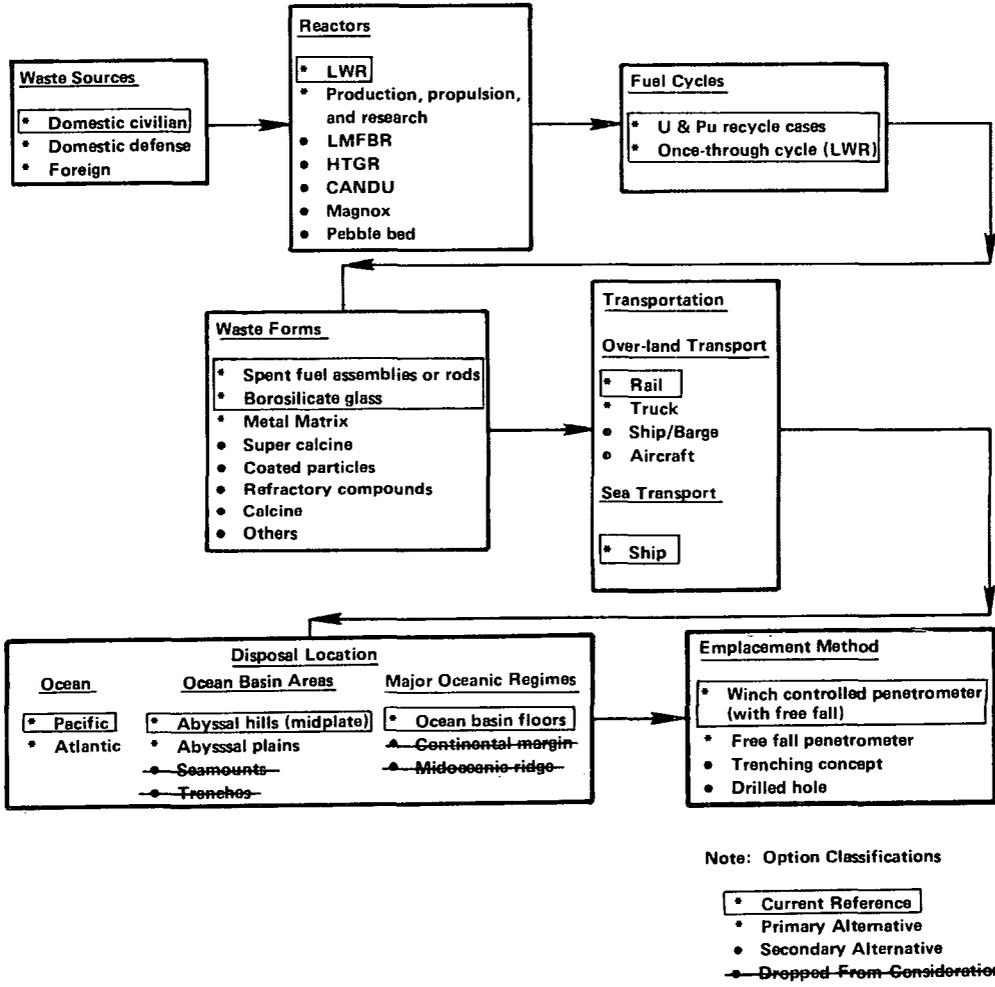


FIGURE 6.1.12. Major Options for the Subseabed Disposal of Nuclear Waste

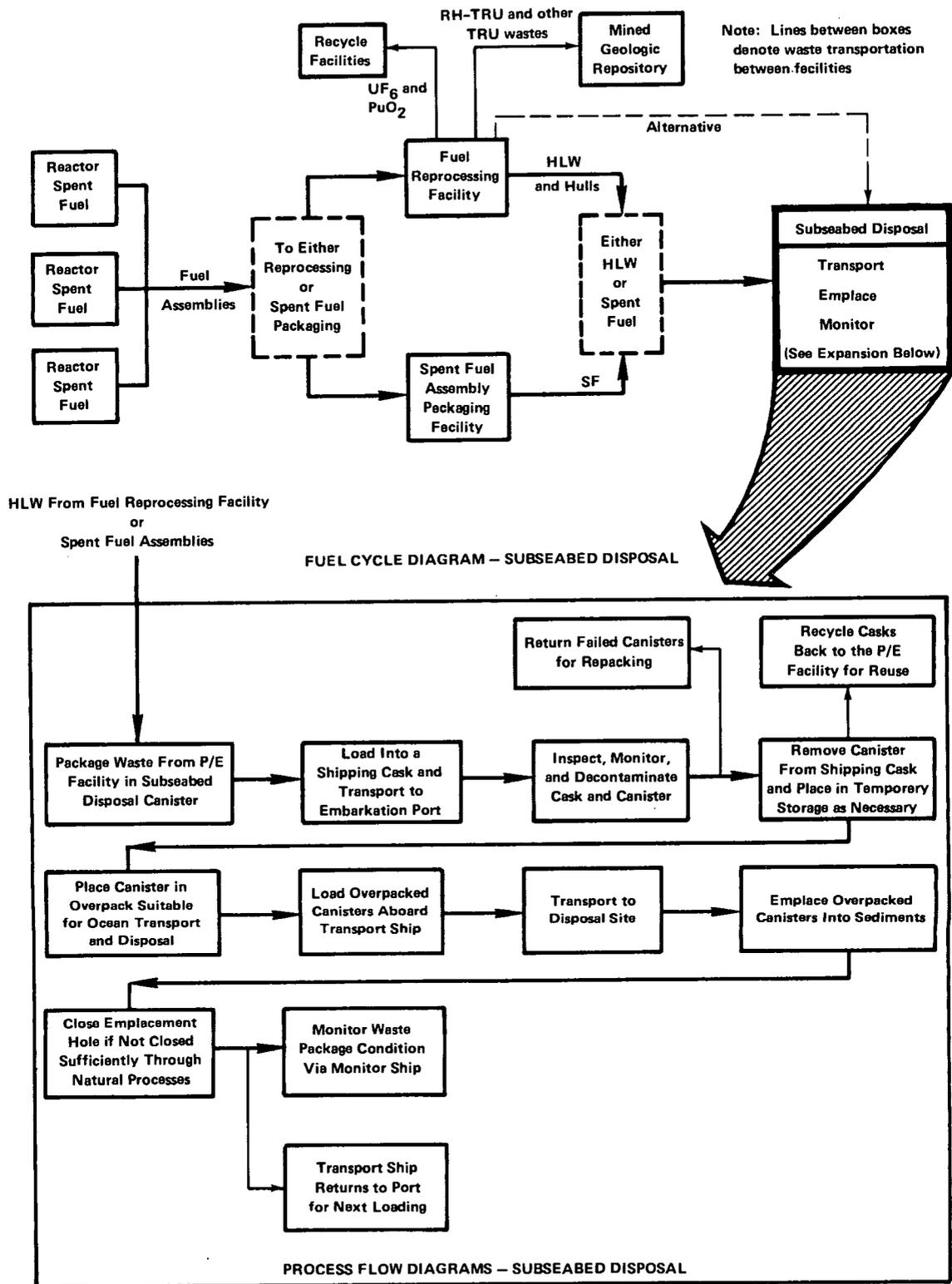


FIGURE 6.1.13. Waste Management System--Subseabed Disposal

The ocean's benthic boundary layer extends from less than 1 m below the sediment-water interface to 100 m above that interface. This layer results from the turbidity induced by natural flow processes and by the biological activity at, or just below, the sediment-water interface. Particulate matter, which would act to sorb radionuclides escaping the sediments, is temporarily suspended in this layer and then returns to the sediment surface.

The water column extends from the benthic boundary layer to the surface of the water. It would provide dilutional mitigation to the release of radionuclides. It would also be a barrier to man's intrusion.

Predisposal Treatment. The predisposal treatment of waste for the subseabed concept would be identical in many respects to the predisposal treatment of waste for the mined geologic repository concept. Chapter 4 of this document discusses the predisposal systems for both spent fuel and HLW common to all of the disposal concept alternatives.

Ocean Environment. Analysis of ocean regimes has shown that the most appropriate areas for subseabed waste containment would be clay-covered abyssal hill regions away from the edges of subocean tectonic plates underlying large ocean-surface currents known as gyres. These vast abyssal hill regions are remote from human activities, have few resources known to man, are relatively biologically unproductive, have weak and variable bottom currents, and are covered with red clay layers hundreds of meters deep.

These clay sediments are soft and pliable near the sediment-water interface and become increasingly rigid with depth. Tests have shown that they have high sorption coefficients (radionuclide retention) and low natural pore-water movement. Surface acoustic profiling indicates that such sediments are uniformly distributed over large areas (tens of thousands of square kilometers) of the ocean floor. As shown by core analysis, they have been continuously deposited and stable for millions of years, giving confidence that they would remain stable long enough for radionuclides to decay to innocuous levels (DOE 1979).

Transportation Features. The overland transportation features of the subseabed disposal concept would be essentially identical to those of the mined geologic disposal concept. In addition, subseabed disposal would require transportation of the waste from the mainland to the subseabed repository. The principal transportation requirements would be for seaport facilities and seagoing vessels.

a. Seaport Facilities. The subseabed reference concept assumes that seaport facilities would be used only for waste disposal activities and would not share services with other commercial endeavors (Bechtel 1979a).

The seaport would have facilities for receiving railway casks containing the waste canisters and for storing them in a water pool until shipment to the repository site. All required handling equipment, including that needed to load the canisters into seagoing vessels, would be available at the port.

The port facility could receive and handle 10,200 spent fuel canisters a year (Bechtel 1979b). For handling high-level reprocessing waste, the total annual throughput would be:

	<u>Canisters</u>
HLW	2,380
Cladding Hulls	2,300
End Fittings	<u>1,520</u>
Total	6,200

Cladding hulls and end fittings are not thermally hot. However, they would be handled in the same manner as HLW for storage and disposal because of their high radiation levels and the possibility of contamination by transuranic elements.

The shipping area of the port facilities would include a canister transfer pool and a transfer cask storage area. To load the ship, the canisters would be moved from the cask and transferred to the ship by crane. The dock facilities would accommodate two ships of the class described below.

b. Seagoing Vessels. Because of the quantities of waste canisters to be disposed of, subseabed disposal would require special dedicated ships (Bechtel 1979a). Each ship would contain equipment for handling the canisters during loading, a water pool to store the canisters during transportation, the necessary equipment to emplace the canisters in the sediment, and water cooling and treatment facilities.

The waste ships could have double hulls and bottoms. Waste canisters would be secured in the holds of the ships in basins filled with water. This concept of transporting fuel canisters in a shipboard storage pool, while new, is considered entirely feasible and is assumed for the reference study.

Disposal of spent fuel might require approximately 15 days to load a ship, 15 days for the round trip from port to repository, and up to 50 days to emplace the canisters at the subseabed site. Thus, a ship would make four trips a year. Based on transporting 1,275 canisters per trip, two ships would be required.

The sea-transportation requirements for HLW would be the same as those for spent fuel assemblies. It is estimated that the same numbers and class of ships as described above would be adequate for transporting HLW and cladding hulls. The same number of trips would be required, but total turnaround time would be about 15 days less because fewer canisters would be handled.

In addition to the ships used for the disposal operations, a survey ship would monitor the emplacement of canisters and their positions relative to one another.

Emplacement. It is assumed that a free-fall penetrometer would provide one alternative method for emplacing canisters in the seabed sediment (Bechtel 1979a). The canisters would have a nose cone to aid penetration and tail fins for guidance. Alternatively, they might be lowered to a predetermined depth and released, and would be designed to penetrate about 30 meters into the sediment. Laboratory tests indicate that the holes made as the canisters entered the sediment would close spontaneously. Canister instrumentation would permit a monitoring crew to track each canister to ensure proper penetration into the sediment and spacing between canisters.

The total seabed area required would be 560 km²/yr (215 mi²/yr) for HLW and 920 km²/yr (354 mi²/yr) for spent fuel assemblies, based on an arbitrary spacing of 300 m (984 ft) between canisters and a waste disposal system of 5,000 MTHM/yr.

Retrievability/Recoverability. Retrievability has not been designed into the system concept (though during the experimental period all emplaced radioactive material would be designed for retrievability) (DOE 1979c). Postemplacement waste-canister recovery from any of the four emplacement options (see Figure 6.1.12) would be possible with existing ocean engineering technology, but estimated costs are high.

Monitoring. After the wastes were emplaced, a monitoring ship would use instrumentation on the ship, on the ocean bottom, and on the canisters to determine information about the buried canister: e.g., its attitude and its temperature. This monitoring would continue for as long as necessary to verify the performance of the sub-seabed isolation system.

6.1.4.3 Status of Technical Development and R&D Needs

Present State of Development

The status of concept design, equipment, and facilities for different facets of a sub-seabed disposal operation is described below.

Emplacement Medium. Properties of the red clay sediment of the ocean's abyssal hills have been studied extensively under the Subseabed Disposal Program (SDP) (Talbert 1977, Sandia 1977, Sandia 1980). The considerable data collected indicate that the sediment is a very promising emplacement medium. The SDP has collected data on nuclide sorption and migration, effects of heat and temperature, ecosystems, and other aspects of the subseabed environment in these sediment areas. The program was started in 1973, and studies of the emplacement medium and of concept feasibility are planned to be completed in 1986. After that, the program would deal with other engineering problems, such as the handling of waste during sea transportation and emplacement (Sandia 1980).

Emplacement Methods. The SDP has not yet defined the methods of waste emplacement in the subseabed. The technical problems associated with this task would be addressed after the studies on sediment properties are completed. In other words, the required depth of emplacement, spacing of canisters, method for assuring hole closure, etc., would have to be known before emplacement methods could be developed.

Four possible methods of emplacement are being considered: (1) free-fall penetrometer, (2) winch-controlled penetrometer descent to a determined depth and final propulsion (the reference concept), (3) trenching, and (4) drilling. The operations are described in Reference 4. The first two methods that use penetrometers present fewer technical challenges since the penetrometer is a widely used tool in marine, land, space, and arctic operations.

Waste Form. The waste form and the canister design required for subseabed disposal of spent fuel have not been determined. Because of the high hydrostatic pressures at the ocean bottom, one important characteristic of the waste package would be a filler material with low compressibility. Generally, metallic fillers would satisfy this requirement, but other solid materials could be more acceptable because of cost advantages, resource conservation, and easier process technology.

The waste form required for storage of HLW in a subseabed repository has not been determined. It is believed that borosilicate glass might be adequate, especially if the temperature of the canister-sediment interface were maintained below 200 C (392 F). This would require adjusting the age of the waste and/or the diameter of the canister to provide rapid heat flow away from the canister. Other waste forms are also being considered.

Waste Containment. Due to the expected effects of high heat and radiation on the properties of the subseabed sediments, waste containment would have to be maintained for a few hundred years to delay the release of nuclides. Experimental data on the rate of corrosion of metallic materials in hot brine and seawater, collected primarily to improve the material performance in desalinization plants and in geothermal applications, would add to the confidence that this capability can be provided.

The SDP has also included laboratory experiments with metallic materials subjected to a seawater environment of 200 C (392 F) and 1,000 psi (6.9×10^6 Pa). Plates of Ticode 12 showed the lowest rate of corrosion, as determined by a weight-loss technique (Talbert 1979).

Facilities. The seaport storage facilities and the facilities that would have to be built aboard ship have not been developed. However, the technology for building them is available since they would resemble existing facilities, such as spent fuel storage pools and ordinary port facilities. The seaport location, size, and capabilities are not yet defined by the SDP.

Technical Issues

The engineering aspects for subseabed disposal have not been established. The transportation logistics, regulations, and the appropriate transportation "package" have not been developed. The precise size and type of facilities that would be built are not known, and the time and motion studies to select the optimum ship size have not been made. In addition, a large area of uncertainty revolves around the methodology that would be used to emplace the waste. Techniques to ensure that waste canisters were placed deep enough into the sediment have not been demonstrated.

If demonstrated, a major attribute of subseabed disposal would be the ability of the sediments to hold radionuclides until they had decayed to innocuous levels. To determine whether these sediments could actually do this, the following technical issues would need resolution.

Ion Transport in the Sediment. More data would be required regarding the rates at which the radioactive ions transfer through the sediment. Studies and empirical data would be required to determine the thermal interaction with canister materials and wastes, conduction, and convection through the sediment.

Ion Transport to the Biosphere. The paths and rates at which the radioactive ions could transfer from the sediment, through the benthic boundary layer, and into the water column are not known. Both mathematical models and empirical experiments would be required to obtain this information. Modeling would also be required to determine a realistic rate of migration up the water column.

Sediment Mechanical Requirements. The subseabed sediments that would be candidates for nuclear waste disposal are between 4,000 and 6,000 m (13,000 and 20,000 ft) below the ocean surface. Further information would have to be acquired regarding their macroscopic (as well as microscopic) structural characteristics. These characteristics include sediment closure after emplacement and long-term sediment deformation and buoyancy resulting from heating.

R&D Requirements

The SDP is divided into seven R&D fields of study (see Sandia 1980), each with numerous subdivisions. As far as funding and the state of technology allow, all of these studies are being pursued simultaneously, though not all at the same level of detail. An eighth field, safeguards and security, would be established later as the results of the other seven studies become known. Brief descriptions of these eight studies which define R&D requirements, follow:

Site Studies. Current studies include evaluation of North Atlantic and North Pacific oceanic areas that meet site suitability criteria. From these areas, certain study locations have been, and will continue to be, identified for more intensified study.

Environmental Studies. Environmental studies include physical and biological oceanography. They focus on analyzing physical characteristics of the water column from the ocean surface to the sediment surface, and on gathering all pertinent information about the marine life that inhabits the water column. The ultimate purpose of these studies is to determine whether, and to what degree, the physical and biological characteristics of the ocean would accelerate or slow the transport of accidentally released radionuclides to man's environment.

Multibarrier Quantification. The multibarrier study includes the sediment, the canister, and the waste form, both immediately adjacent to the waste container and further afield, to determine their natural characteristics. Again, the ultimate purpose is to learn whether, and to what degree, they would allow released radionuclides to be transported. A second purpose is to learn how they would react to the heat and radiation generated by a waste container, as well as to any engineered modification to the sediment such as artificial closing of the emplacement hole.

Transportation. Transportation studies include four subdivisions:

- Land transport with investigations directed to transporting HLW and/or spent fuel from an originating plant to the port facility by rail, road, or barge.
- The port facility, including a receiving structure.
- The staging area, to include cooling facilities for holding waste packages until they could be loaded.
- Sea transport with studies including design of special transport/emplacement vessels and of travel routes designed to minimize interaction with shipping lanes and all other forms of maritime activity. It is likely that this would be a self-powered ship, but it could be a vessel that could be towed, possibly under water. Transportation technology is in early planning stages, pending determination of disposal feasibility.

Emplacement and Monitoring. The study of emplacement and monitoring focuses on the time period that begins when waste packages would be removed from their cooling area on the transport vessel and continues through burial deep in the subocean sediments and closure of the entrance hole, either naturally or artificially. An intrinsic part of the process would be the monitoring function. Monitoring would include surveying precise disposal locations, guiding emplacement mechanisms into those locations, and tracking the integrity, attitude, and stability of waste containers for as long as would be required after emplacement.

Social/Political Studies. Even if technological and environmental feasibility for the subseabed disposal concept were established, domestic and international institutions would ultimately determine whether the concept could be used. There are no laws or agreements at this time that specifically prohibit or allow subseabed disposal. Issues important to this area are further discussed in Section 6.1.4.4 under International and Domestic Legal and Institutional Considerations. International agreements and structures would enhance the implementation of the concept. Evaluation of the current political and legal postures of all countries that might be involved in subseabed disposal is under way. The existence of an international NEA/OECD Seabed Working Group is indicative of the international interest in the concept.

Risk/Safety Analyses. As data become available, risk and safety analyses would be completed on all aspects of the SDP.

Security and Safeguards. Except in the most general terms, studies in these areas would have to await data acquisition and assessment.

R&D Costs/Implementation Time

Research and development is assumed to end when the technology had been translated into routine practice at the first facility. Follow-on R&D in support of facility operation is considered in a different category.

To date, almost all resource expenditures have been focused on the technical and environmental feasibility of the subseabed geologic concept, rather than on specific on-site studies or demonstrations of current engineering practice. The estimated total R&D costs are \$250 million (DOE, 1979).

The SDP program plan has been divided into four distinct phases (Sandia, 1980). In each phase, the concept feasibility is assessed. The estimated completion dates shown do not consider programmatic perturbances resulting from regulatory or institutional influences.

- Phase 1 Estimation of technical and environmental feasibility on the basis of historical data. Completed in 1976.
- Phase 2 Determination of technical and environmental feasibility from newly acquired oceanographic and effects data. Estimated completion date: 1986.
- Phase 3 Determination of engineering feasibility and legal and political acceptability. Estimated completion date: 1993-95.
- Phase 4 Demonstration of disposal facilities. Estimated completion date: 2000 to 2010 (Anderson et al. 1980).

Summary

Major uncertainties, shortcomings, and advantages of the concept are summarized below:

- The remoteness of the location, apparent sorption capacity of the sediments, and demonstrated stability of the site are attractive attributes.
- The concept could be implemented in a step-wise fashion.
- The expected performance of packages and waste form in the environment at the seabed is not well understood.
- Specific new domestic legislation and international agreement would likely be required.
- Retrievability to allow for corrective action purposes might be difficult.
- Transportation requirements to a remote location add to the overall risk of the concept.

6.1.4.4 Impacts of Construction and Operation (Preplacement)

Health Impacts

Both radiological and nonradiological health impacts are discussed below.

Radiological Impacts. Both occupational and nonoccupational doses prior to the waste arriving at the seaport facility are expected to be similar to those anticipated for a mined geologic repository, as presented in Chapters 4 and 5.

The occupational and nonoccupational radiological impacts of the operation of the seaport facility and the seagoing vessels have been developed by Bechtel (1979a), and are presented in Table 6.1.11. These impacts are conservatively estimated as equivalent to those for away-from-reactor storage pools (AFR), corrected in consideration that:

- The primary waste handled at the subseabed facilities would be 10 years old.
- The primary waste at the subseabed facilities would be encapsulated.
- The number of personnel is expected to be smaller at the seaport facility than at the AFR facility. This may be offset by the fact that personnel might receive occupational doses for longer time periods while serving aboard ship.

TABLE 6.1.11. Radiological Impacts Of The Normal Operation
At A Subseabed Repository

	Whole Body Dose, man-rem/yr	
	Spent Fuel	High-Level Waste
Occupational		
Seaport Facility	340	200
Seagoing Vessels	340	200
Nonoccupational		
Seaport Facility	40	10
Seagoing Vessels	Negligible	Negligible

Bechtel (1979a) gives the consequences of abnormal events at subseabed facilities. These consequences are equated with accidents postulated for the AFR (i.e., design basis tornado) facility for the most exposed public individual. No probability analysis was included. For spent fuel disposal, the radiological impacts of an abnormal event would be 0.02 mrem/event for the seaport facility and 0.003 mrem/event for the seagoing vessels. For HLW, these impacts would be 0.001 mrem/event and 0.002 mrem/event, respectively.

The maximum risk would be posed by the sinking of the seagoing vessel or by loss of waste canisters overboard. Except for accidents in coastal waters where mitigation actions could be taken, the radioactive materials released into the sea following such an event would disperse into a large volume of the ocean. Some radionuclides might be reconcentrated through the food chain to fish and invertebrates, which could be eaten by man. Bechtel (1979a) assumes that the waste could be retrieved if either event were to occur and does not provide an impact estimate. The doses provided in Table 6.1.12 for such an event are taken from EPA (1979).

Nonradiological Impacts. The numbers of injuries, illnesses, and deaths related to the construction and operation of the subseabed disposal option prior to the waste arriving at the seaport facility/repository are expected to be similar to those for the mined geologic options. At the seaport facility, it is estimated that the impacts would be no greater than those associated with surface storage and transfer facilities to be used with a reprocessing plant or spent fuel overpacking facility. These impacts are discussed in Chapter 4.

Additional areas specific to subseabed disposal that would have nonradiological health impacts are the construction of seagoing vessels and the conduct of operations at a seaport and on the ocean. Although there are no quantitative estimates of these impacts, it is anticipated that they would be similar to those incurred during the construction and operation of conventional seagoing vessels and operation of conventional dock facilities.

Natural System Impacts

Impacts to the natural environment for this disposal option would be related primarily to transportation and emplacement activities. Radiological concerns would be most significant

TABLE 6.1.12. Estimated Dose Commitment From Marine Food Chain For Loss of Waste At Sea

	<u>Population</u> <u>man-rem</u>	<u>Average Individual,</u> <u>rem</u>
Undamaged Spent Fuel		
Continental Shelf	510	5.9×10^{-4}
Deep Ocean	100	1.1×10^{-4}
Damaged Spent Fuel		
Continental Shelf	1×10^5	0.11
Deep Ocean	100	1.1×10^{-4}
HLW (Plutonium Package)		
Continental Shelf	Not provided	Not provided
Deep Ocean	100	1.1×10^{-4}

under abnormal conditions, while nonradiological impacts could also pose problems under normal operating conditions.

Transportation-related impacts for those activities occurring before the waste material was loaded on the ships would be similar to those for a mined geologic repository. Once the material was loaded onto the ships, impacts to the marine environment would have to be considered. In the case of potential accident conditions at sea, the design of the waste transporting vessels to include double hulls and bottoms would reduce the likelihood of releasing harmful material into the environment.

There are several uncertainties that limit the ability to predict natural system impact levels with confidence. Of primary concern is a lack of understanding of ion transport within the sediment and biosphere, including the benthic region, the water column and ocean life forms. In addition, the extent of the isolation barrier that the resealed sediment would provide after emplacement is not clear. Each of these factors makes detailed impact assessment difficult.

Other subseabed disposal impacts identified, but not quantified by Bechtel (1979a), include minor air emissions from construction equipment, dust generation, and road, rail, and vessel emissions. Construction-related impacts on water quality and vegetation as well as impacts on the marine environment resulting from dredging and breakwater construction could be locally significant. Although these impacts were identified by Bechtel (1979a), there are no data that indicate they would be significant.

Socioeconomic Impacts

Because a major land repository would not be required under this option, the most important socioeconomic impacts would be attributable to transportation activities. Transportation activities fall into three categories: (1) transportation of wastes on land to the port where the wastes would be transferred to the ship, (2) waste-handling activities at the port facility, and (3) ocean transportation from the port facility to the point where the material would be deposited in the seabed sediment.

Socioeconomic impacts would be concentrated at the point where support activities were most intense: at the port facility. The nature of the activity has led certain reviewers to conclude that one of the most significant factors associated with this disposal option would be difficulty in finding a suitable dedicated (Bechtel 1979a). Moreover, they project moderate community impacts and suggest that local socioeconomic impacts could reach significant levels.

Detailed projections of the impact of implementing this disposal option on the public and private sectors could be made only on site-specific basis. Nevertheless, impacts would be expected in the coastal area near the port facility. The total anticipated increase in employment for a 5000 MTHM per year disposal system, although quite concentrated, is expected to be less than 2000 people.

Aesthetic Impacts

The significance of aesthetic impacts would depend on the appearance and operating parameters of a facility, as well as on the extent to which it would be perceived by humans. For the subseabed disposal option, much of the waste-handling and transportation activities would occur in remote areas of the ocean. Consequently, the aesthetic impacts, regardless of their nature, would not be significant.

Aesthetic impacts near the port facility, however, could be locally significant. Such impacts could be accurately determined only on a site-specific basis. However, it is important to recognize that the required port facilities for a nuclear waste handling facility would be substantial.

Resource Consumption

Use of energy and construction of seaport facilities and seagoing vessels would be the primary resource consuming activities in this option. Energy would be consumed during land transportation, loading, and sea transportation activities. A quantitative estimate of energy consumption is provided in Table 6.1.13.

The seaports would have facilities for receiving railway casks containing the waste canisters and for placing them in interim storage. Interim storage pools should be able to handle one-half of the anticipated yearly volume of wastes (2500 MTHM) and are expected to

TABLE 6.1.13. Estimated Energy Consumption

	Spent Fuel	HLW
Propane m ³	2.4 x 10 ⁴	1.0 x 10 ⁷
Diesel, m ³	5.0 x 10 ⁶	1.6 x 10 ⁶
Electricity, KWh	2.0 x 10 ¹⁰	5.7 x 10 ¹⁰

require an area within the boundaries of the port area subseabed support facilities of 2320 m² (25,000 ft²) (Bechtel 1979a). Other storage and transfer facilities would also be needed. The total area required for all the required facilities is expected to be over 3600 ha (8500 acres).

Construction of the waste disposal ships with double hulls and bottoms, waste handling equipment for loading, and carefully constructed compartments for holding the wastes during transportation activities, like construction of the port facilities, would lead to the consumption of steel and other basic construction materials. An estimate of the material consumption is provided in Table 6.1.14.

International and Domestic Legal and Institutional Considerations

The subseabed disposal option, like the island and ice sheet options, would require transporting waste material over the ocean, and the general international implications of such transportation are important.

Any implementation of subseabed disposal is far enough in the future that many current legal and political trends could change. However, it is not too early to identify important problems, so that possible developments could be foreseen and controlled.

The use of subseabed disposal would be governed by a complex network of legal jurisdictions and activities on both national and international levels. Domestic use of subseabed disposal of radioactive waste would require amendment of the U.S. Marine Protection, Research, and Sanctuaries Act of 1972 (The Ocean Dumping Act) which currently precludes issuance of a permit for ocean dumping of high-level radioactive waste.

Table 6.1.14. Estimated Material Consumption for Ship and Facility Construction (in MT)

	Spent Fuel	HLW
Carbon Steel	877,000	282,000
Stainless Steel	83,500	22,500
Components		
Chromium	14,200	4,600
Nickel	7,500	2,000
Tungsten	---	---
Copper	1,400	1,900
Lead	12,900	2,900
Zinc	1,200	600
Aluminum	13,000	1,400

The London Convention of 1972, a multinational treaty on ocean disposal, addresses the problem of dumping of low-level and TRU wastes at sea and bans the sea dumping of high-level

wastes (Deese 1976). This treaty is currently being revised to deal more specifically and completely with the problem of dumping low-level and some TRU wastes. This treaty arguably does not preclude the controlled emplacement of high-level wastes or spent fuel into geologic formations beneath the ocean floor. However, the intended prohibition of the treaty would require clarification.

Subseabed disposal might offer the important political advantage of not directly impacting any nation, state, or locality. Likewise, the alternative might have the disadvantage of incurring risk to nations that do not realize the benefits of nuclear power generation.

Assuming that the real impact uncertainties associated with the subseabed concept were resolved, the primary political disadvantage of subseabed disposal would be its possible perception as an ecological threat to the oceans. If publics, governments, and international agencies were to view such disposal as merely an extension of past ocean dumping practices, implementation would be difficult if not impossible. However, if this option were understood as involving disposal in submarine geologic formations that have protective capacities comparable to or greater than similar formations on land, opposition might be less.

6.1.4.5 Potential Impacts Over the Long Term (Postemplacement)

Potential Events

Earthquakes, volcanic action, major climatological and circulatory changes, and meteorite impacts are examples of natural processes that might affect subseabed containment stability. Careful selection of the ocean area would minimize the probability of the first three events occurring. There is no known method of minimizing the probability of meteorite impact other than concentrating emplacement, which, while reducing the random target area, would correspondingly increase the potential consequences if a meteorite did strike. On the other hand, other damage caused by any meteorite that could penetrate 5 km (3 mi) of water would make the release of emplaced radioactive waste insignificant.

For HLW disposed of in a subseabed repository, a very low probability for criticality is assumed because of the great distances between canisters at the bottom of the sea. For spent fuel, the probability of criticality might be somewhat greater because of the higher fissile content of a single canister.

Since the site would be located in a part of the ocean with no known materials of value, future human penetration would be highly unlikely.

Potential Impacts

Two models have been developed by Grimwood and Webb (1976) to characterize the physical transport and mixing processes in the ocean, as well as incorporation in marine

food chains and ultimate consumption of seafood and radiation exposures to man. Although there is some question as to the applicability of these models to the subseabed disposal option, the following summary of results using these models is presented until such time as better estimates of radiation exposures to man from subseabed disposal are available.

The individual doses resulting from the consumption of surface fish, deep-ocean fish, or plankton are expected to be well below the maximum permissible levels. External individual doses^(a) from contamination of coastal sediments are expected to be fractions of the ICRP dose limit for both skin and whole body irradiation. The largest annual internal population doses to the whole body and bone due to the consumption of surface fish would be about 4×10^4 and 10^5 man-rems, respectively. The largest annual external population doses from contaminated sediments would be about 10^3 to 10^4 man-rems for both skin and whole body. These large population doses would occur during the early stages of postemplacement and would decrease during the later stages.

As an attempt to provide a further yardstick against which to compare the results of the calculations, Table 6.1.15 gives the concentrations of nuclides predicted by the modeling, as well as the natural activity in seawater.

6.1.4.6 Cost Analysis

An estimate of capital, operating, and decommissioning costs for subseabed disposal has been made for both spent fuel disposal and HLW disposal (Bechtel 1979a). Both are based on penetrometer emplacement. All estimated costs are in January 1978 dollars.

TABLE 6.1.15. Levels Of Natural And Wastes Radionuclides In Seawater

Nuclide	Natural Activity In Seawater, Ci/cm ³	Max Widespread Surface Water Conc. Predicted From Postulated Waste Disposal Operation, Ci/cm ³ (No Containment)
Actinides		
Pb-210	(1 - 9) $\times 10^{-11}$	2×10^{-15}
Pb-210	1×10^{-10}	2×10^{-15}
Ra-226	1×10^{-10}	2×10^{-15}
Th-230	(0.6 - 14) $\times 10^{-13}$	2×10^{-17}
Th-234	1×10^{-9}	1×10^{-15}
U-234	1×10^{-9}	1×10^{-15}
U-238	1×10^{-9}	4×10^{-15}
Pu-239		1×10^{-12}
Fission Products		
H-3	2×10^{-10}	1×10^{-12}
Sr-90		4×10^{-10}
I-129	3×10^{-11}	3×10^{-14}
Cs-137		6×10^{-10}

(a) Based on world population

In each case, only those costs associated with and peculiar to subseabed disposal are addressed. Facilities common to all disposal options under consideration, such as transportation and geologic repository facilities, are not specifically addressed.

Capital Costs

The capital costs for the subseabed disposal alternative are categorized as follows.

Seaport Interim Storage Facility. This installation would provide receiving facilities for 5,000 MTHM/yr of spent fuel assemblies in 10,200 canisters. It would also be designed to provide interim storage for 5,000 canisters (2,500 MTHM). The same facility would receive the HLW and hulls from a 5,000 MTHM/yr fuel recycling system. Interim storage would be provided for 3,100 of these canisters at the port facility.

The seaport interim storage facility would be similar to a packaged fuel receiving and interim storage facility (Bechtel 1977) appropriately adjusted for size and waste form. The capital cost estimates are \$240 million for the spent fuel case and \$190 million for the HLW case.

Port Facility. The port facilities for both disposal cases are assumed to be identical for cost estimating purposes. The capital cost estimate is based on a recent estimate of another facility (Bechtel 1979a). The estimate for this port is \$24 million.

Disposal Ships. The two disposal ships for the spent fuel case would have a capacity of 1,275 canisters each, while those for the HLW case would have a capacity of 775 canisters each. Since the canister capacity difference would be offset by the heat load and cooling requirement difference, the ships are assumed to be identical for estimating purposes.

The capital cost estimate of the ships is based on an estimate for a mining ship (Global Marine Development, Inc. 1979) appropriately adjusted. The estimated capital cost of the two disposal ships is \$310 million (\$155 million each). Note however that sophisticated offshore oil well drilling ships have been reported to cost between \$50 million and \$70 million each (Compass Publications 1980) or about half the above estimate.

Monitoring Ship. The capital cost for the monitoring ship was estimated from available data for oceanographic vessels. The estimate is \$3.0 million for the ship and an additional \$0.9 million for navigation and control, special electronics, and other surveillance equipment and for owner's costs. This brings the total capital cost to \$3.9 million (Treadwell and Keller 1978).

Operating Costs

Operating costs for the subseabed disposal concept are estimated on a per year basis based on 5,000 MTHM/yr of both waste forms (spent fuel and HLW). This would result in virtually the same sea transportation requirements (number of trips per year). However, differences would occur for the HLW disposal case in years 1 through 9, when only hulls would be

processed and disposed of, and during years 41 through 49, when only HLW would be disposed of.

The estimated yearly operating costs for the subseabed disposal concepts are presented in Table 6.1.16.

Operating costs associated with the reference subseabed disposal concept but also common to other disposal concepts are assumed to be similar. These costs would include transportation, AFR facilities (for the spent fuel), P/E facilities, and geologic repository facilities (assumed for the reference concept).

Decommissioning Costs

Decommissioning costs particularly associated with subseabed waste disposal operations would probably be limited to the seaport, interim storage facility, the port facility, and the disposal ships. The monitoring ship is not expected to be affected by radioactive waste during its 40 years of operation. Any decommissioning costs associated with the monitoring ship are assumed to be offset by its salvage value, which results in a zero net decommissioning cost.

The decommissioning cost of an AFR facility is used as the basis for the decommissioning cost of the seaport interim storage facility (Bechtel 1979b). These costs, based on 10 percent of capital cost excluding owner's cost, are approximately \$23 million for the spent fuel disposal and approximately \$18 million for the HLW disposal case.

The decommissioning costs for the port facility and two disposal ships are the same for both waste forms and are estimated to be about \$2 million and \$29 million, respectively, assuming 10 percent of capital cost less owner's costs.

Costs for decommissioning other facilities associated with subseabed disposal and common to other waste disposal alternatives are assumed to be similar. These facilities include AFR facilities (for the spent fuel), P/E facilities, and geologic repository facilities. These

TABLE 6.1.16. Estimated Operating Costs

Facility	Estimated Cost, \$ million/yr	
	Spent Fuel Disposal	HLW Disposal
Seaport Interim Storage Facility		
Years 1-9	---	3.4
Years 10-40	6.2	4.9
Years 41-49	6.2	3.4
Port Facility	1.5	1.5
Disposal and Monitoring Ships		
Years 1-9	---	14.5
Years 10-40	20.9	20.9
Years 41-49	20.9	14.3

total costs are estimated to be about \$398 million for the spent fuel disposal and \$721 million for the HLW disposal.

6.1.4.7 Safeguard Requirements

Because this concept may involve both subseabed and mined geologic disposal, its implementation could require safeguarding two separate disposal paths. The risk of diversion for the subseabed disposal concept would be primarily a short-term concern because of the remoteness of the disposal site and the major operational and equipment requirements that would have to be satisfied for retrieval. Physical protection of the sensitive facilities and transportation operations would be the most effective way to deny access to the waste for the short term, as is common to most waste disposal concepts. See Section 4.10 for additional discussion of predisposal operations safeguards requirements.

6.1.5 Ice Sheet Disposal

6.1.5.1 Concept Summary

It is estimated that, without significant climatic changes, the continental ice sheets could provide adequate isolation of high-level radioactive waste from the earth's biosphere. However, the long-term containment capabilities of ice sheets are uncertain. Areas of uncertainty have been reviewed by glaciologists (Philberth 1958, Zeller et al. 1973, and Philberth 1975). These reviewers cited the advantages of disposal in a cold, remote, internationally held area and in a medium that should isolate the wastes from man for many thousands of years to permit decay of the radioactive components. But they concluded that, before ice sheets can be considered for waste disposal applications, further investigation is needed on:

- Evolutionary processes in ice sheets
- Impact of future climatic changes on the stability and size of ice sheets.

Most of the analysis in these studies specifically addresses the emplacement of waste in either Antarctica or the Greenland ice cap. Neither site is currently available for waste disposal for U.S. programs: Antarctica because of international treaties and Greenland because it is Danish territory.

Proposals for ice sheet disposal suggest three emplacement concepts:

- Meltdown - emplaced in a shallow hole, the waste canister would melt its own way to the bottom of the ice sheet
- Anchored emplacement - similar to meltdown, but an anchored cable would allow retrieval of the canister
- Surface storage - storage facility would be supported above the ice sheet surface with eventual slow melting into the sheet.

Ice sheet disposal, regardless of the emplacement concept, would have the advantages of remoteness, low temperatures, and isolating effects of the ice. On the other hand, transportation and operational costs would be high, ice dynamics are uncertain, and adverse global climatic effects are a possibility.

6.1.5.2 System and Facility Description

Systems Options

The reference concept for the initial ice sheet disposal of nuclear waste has been developed from a number of options available at each step from the reactor to disposal in the ice sheet. It includes the three basic emplacement options and was selected through judgment of a "most likely" approach based on available information and is not supported by a detailed system engineering analysis.

Various options to be considered for ice sheet disposal are summarized in Figure 6.1.14. The bases for selection of the options chosen for the reference design (those blocked off) are detailed in a variety of source material cited in Appendix M.

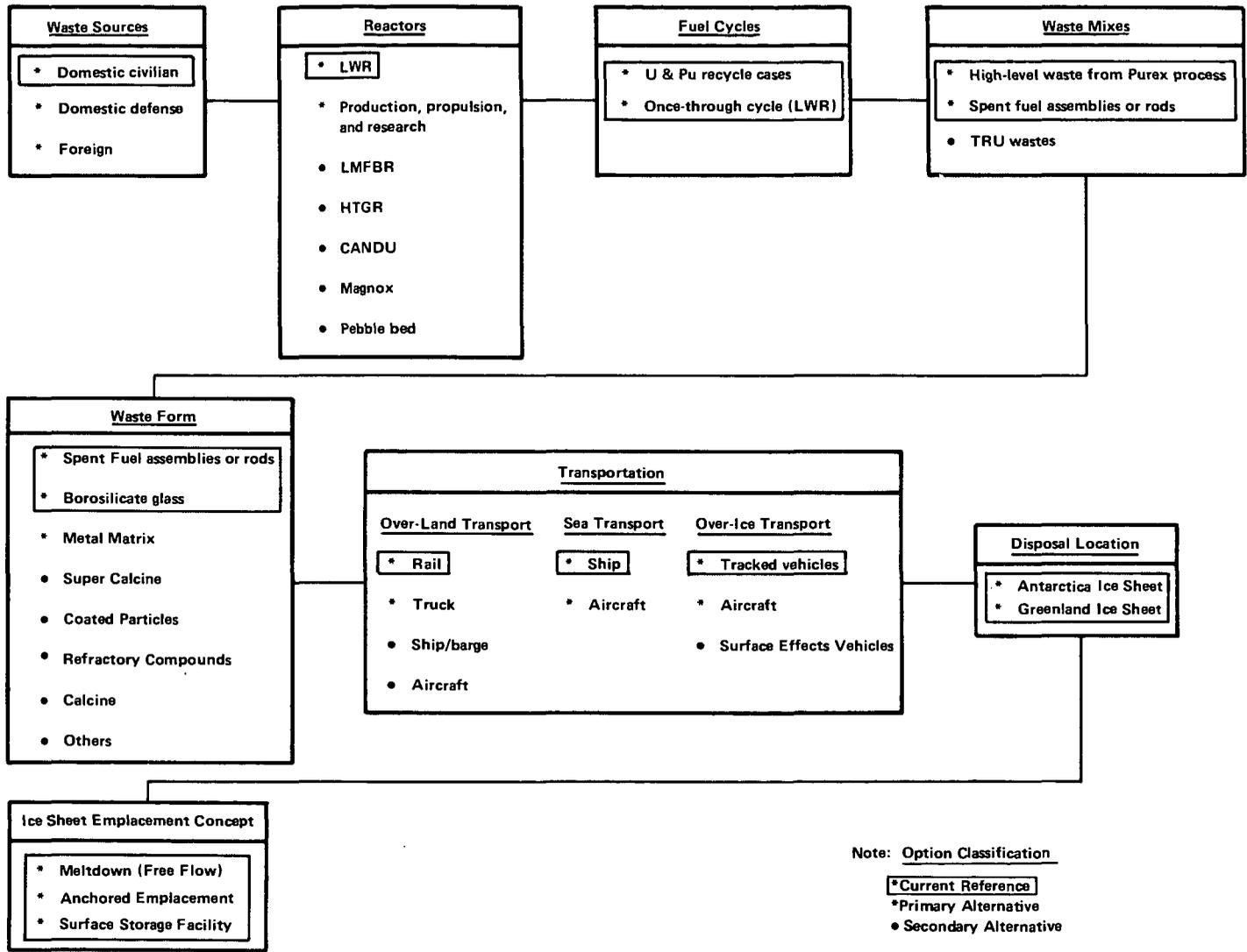


FIGURE 6.1.14. Major Options for Ice Sheet Disposal of Nuclear Waste

Because the options for the waste disposal steps from the reactor up to, but not including, the transportation alternatives are similar to those for a deep geologic repository, the options selected for the reference design are similar for the two concepts. From that point on, the options selected for the reference ice sheet design are based on current program documentation for ice sheet disposal.

Waste-Type Compatability

Ice sheet disposal by meltdown has been considered primarily for solidified, high-level wastes from nuclear fuels reprocessing. It would also be applicable for direct disposal of spent fuel, without reprocessing, although meltdown would be marginal if the fuel were emplaced 2 years after reactor discharge. The feasibility of meltdown emplacement of cladding hulls and fuel assembly hardware is questionable because the canister heating rate from radioactive decay would be less than 1/10 that in HLW waste canisters.

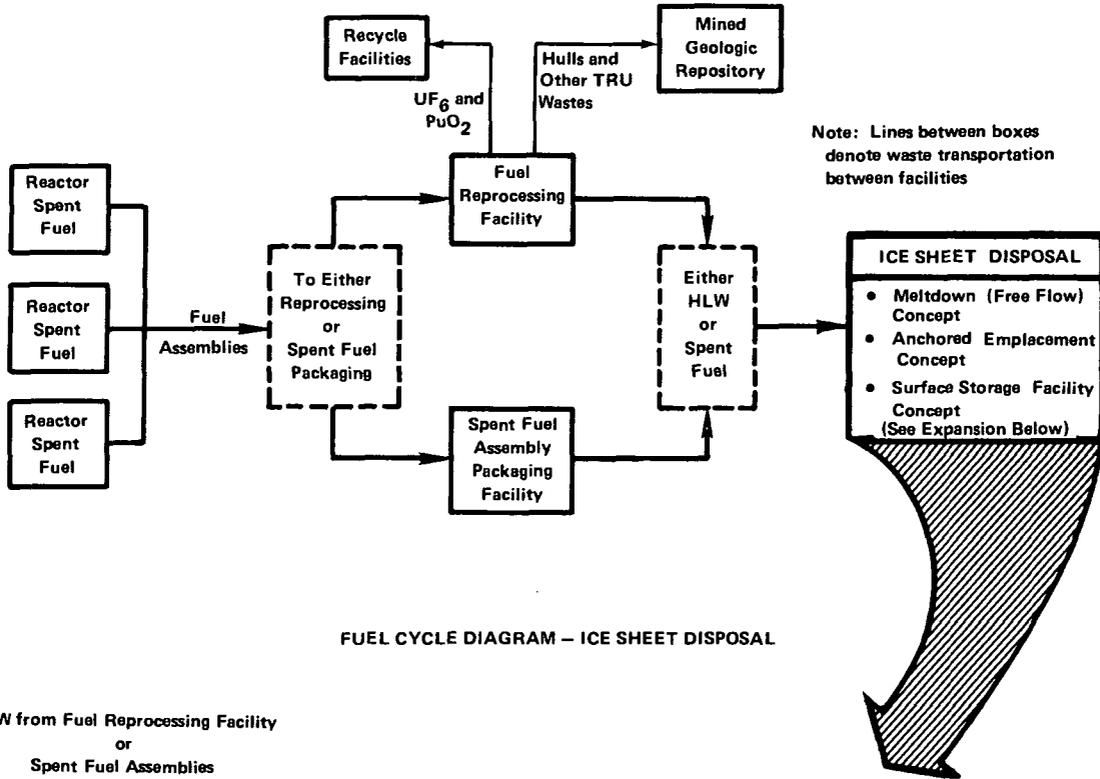
For most TRU waste, the heating rate would be less than 1/1000 that expected in HLW waste canisters, and the meltdown concept does not appear to be feasible. Without blending with HLW, disposal of this waste would be limited to storage in surface facilities on the ice or emplacement in shallow holes in the ice. For these options, the waste would be buried gradually in the ice sheet. Contact handled and remotely handled TRU wastes could be handled in a similar manner. Because of volume and cost considerations, TRU wastes are assumed to be placed in other terrestrial repositories.

Waste System Description

The ice sheet waste management system is detailed in Figure 6.1.15. This system concept is very similar to the very deep hole concept since both spent fuel and the uranium-plutonium recycle cases could be treated and mined geologic repositories could augment disposal.

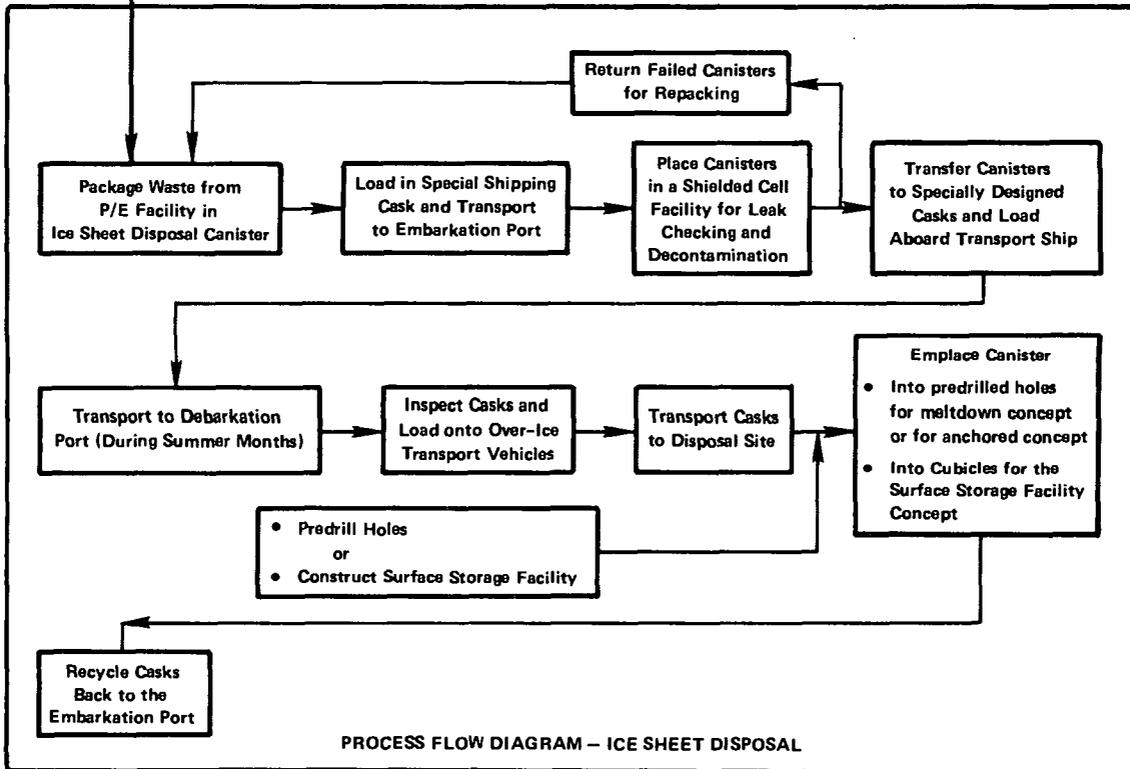
The reference ice sheet disposal concept is not yet well defined. None of the three basic emplacement concept alternatives proposed in the literature (Battelle 1974, EPA 1979, and ERDA 1976) has been selected as a reference or preferred alternative. Waste disposal by any one of these three concepts would be either in the Antarctica or Greenland ice sheets. A generalized schematic of the waste management operational requirements is provided in Figure 6.1.16 (Battelle 1974). The schematic shows the basic system operations (EPA 1979):

- Predisposal treatment and packaging at the reprocessing plant
- Transporting solidified waste from the reprocessing plant or interim retrievable surface storage facility by truck, rail, or barge to embarkation ports
- Marine transport by specially designed ships during 1 to 3-month periods of each year.
- Unloading the waste canisters at a debarkation facility near the edge of the land mass
- Transporting over ice by special surface vehicles or aircraft on a year-round basis, as practicable
- Unloading and emplacing the waste canisters at the disposal site.



FUEL CYCLE DIAGRAM - ICE SHEET DISPOSAL

HLW from Fuel Reprocessing Facility
or
Spent Fuel Assemblies



PROCESS FLOW DIAGRAM - ICE SHEET DISPOSAL

FIGURE 6.1.15. Waste Management System--Ice Sheet Disposal

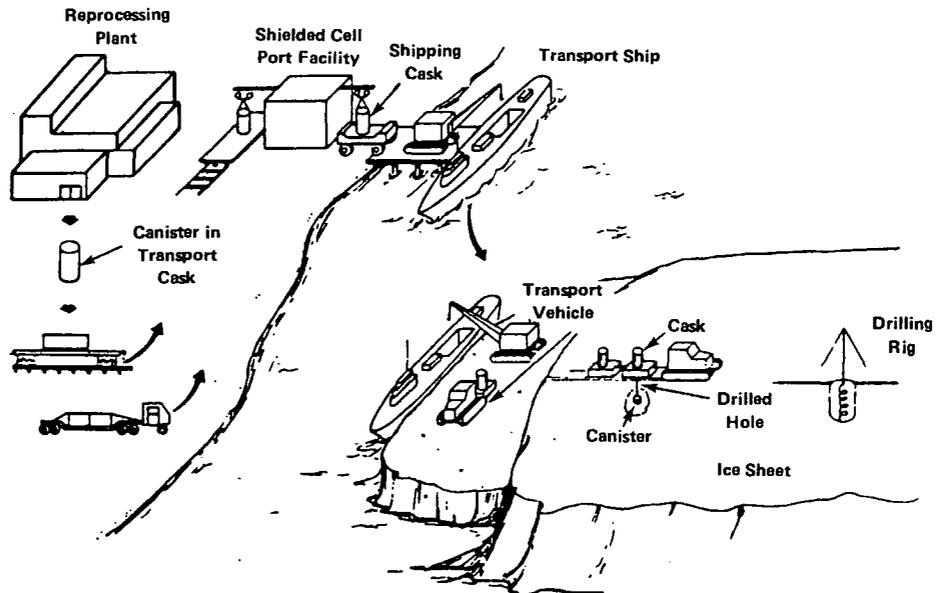


FIGURE 6.1.16. Schematic of Operations in Ice Sheet Disposal Systems for High-Level Radioactive Wastes (18)

Predisposal Treatment and Packaging. The predisposal treatment of waste for the ice sheet concept would be identical in many respects to the predisposal treatment of waste for the mined geologic repository concept. Chapter 4 discusses the predisposal systems for both spent fuel and HLW common to all the various alternative concepts for waste disposal.

Transportation and Handling. Transportation to the disposal site would probably be accomplished in three steps, as indicated above. First, all the waste canisters would be loaded into heavily shielded transport casks for shipment from the interim storage site to the embarkation port. Waste containers would accumulate at the embarkation port in the U.S. on a year-round schedule. There, the canisters would be unloaded in a shielded cell facility and examined for leakage, contamination, damage, or other unsuitable conditions. The canisters would be overpacked, transferred individually to specially designed casks, and loaded aboard a specially designed transport ship for shipment to the ice sheet. Acceptable canisters could also be stored for up to a year in an interim retrievable surface storage facility (Szulinski 1973). Any unacceptable canister would either be corrected on site or returned to the reprocessing plant or another appropriate handling facility.

Landing and discharge operations at the ice sheet would require special facilities and would be limited to the summer months. At the debarkation port, the casks would be inspected and unloaded onto over-ice transport vehicles. After transport to the disposal site, the canisters would be lowered from the casks to the emplacement site and the casks would be recycled back to the embarkation port. An alternative transportation mode would be to fly the waste canisters from the debarkation site to the emplacement site.

It appears possible, as an alternative, that the same shipping cask might be used for handling a waste canister first at the reprocessing plant, then for marine transport to the ice sheet, and finally for over-ice transport to the disposal site.

Debarkation ports on the ice sheets with handling systems for unloading casks directly onto the over-ice transport system would be possible in the Antarctic or in Greenland, but might be very expensive. The currently preferred alternative is to dock the transport ship at a land-based port in an ice-free area to unload the casks into the over-ice transport vehicles.

Emplacement. The waste canisters would be disposed of using one of the three basic concepts described in detail below.

The melt-down or free flow concept is shown in Figure 6.1.17 (ERDA 1976). Waste would be disposed of by selecting a suitable location in the ice sheets, predrilling a shallow hole, lowering the canister into the hole, and allowing it to melt down or free flow to the ice sheet base and bedrock beneath (EPA 1979).

The surface holes would be predrilled to depths from 50 to 100m and would provide protective shielding from radiation during canister emplacement. To avoid individual canisters interfering with each other during descent and possible concentration at the ice sheet base, the suggested spacing between holes is about 1000 m.

The canister meltdown rate is based on calculations from the penetration rates of thermal ice probes. It is estimated that the rate of descent for each canister would be on the order of 1.0 to 1.5 m/day. Assuming only vertical movement and an ice sheet 3000 m (9900 ft) thick, meltdown to the bedrock would take 5 to 10 years.

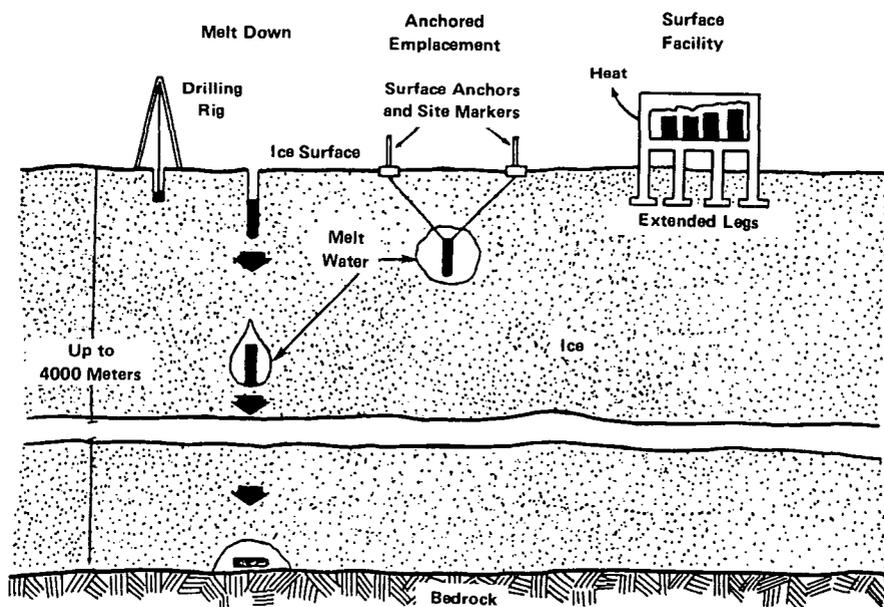


FIGURE 6.1.17. Ice Sheet Emplacement Concepts

An important factor in this concept would be the design and shape of the canister, which should help assure a vertical path from surface to bedrock. In addition to the canister design and shape, the type of construction materials would be important. Specifications for these materials would have to include consideration of differences in ice sheet pressure and the possibility of saline water at the ice/ground interface. A multibarrier approach that gives consideration to the total waste package and its emplacement environment would be required. This approach would be equally applicable to the anchored emplacement and surface storage alternatives.

The anchored emplacement concept, also shown in Figure 6.1.17, would require technology similar to that required by the meltdown or free flow concept described above, the difference being that this concept would allow for interim retrieval of the waste (EPA 1979). Here, cables 200 to 500 m (660 to 1650 ft) long would be attached to the canister before lowering it into the ice sheet. After emplacement the canister would be anchored at a depth corresponding to cable length by anchor plates on or near the surface. The advantage over the meltdown concept is that instrument leads attached to the lead cable could be used to monitor the condition of the canister after emplacement.

Following emplacement, new snow and ice accumulating on the surface would eventually cover the anchor markers and present difficulties in recovery of the canister. The average height of snow and ice accumulating in the Antarctic and Greenland is about 5 to 10 cm/yr (2 to 4 in./yr) and 20 cm/yr (8 in./yr), respectively. However, climatic changes might result in a reversal of this accumulation with ice being removed from the surface by erosion or sublimation. If continued for a long period of time such ice surface losses could expose the wastes. Recovery of canisters 200 to 400 years after emplacement might be possible by using 20-m (66-ft)-high anchor markers. It would take about 30,000 years for the entire system to reach ice/ground interface at a typical site. During that time, the canisters and anchors would tend to follow the flow pattern of the ice (Battelle 1974).

The surface storage facility concept would require the use of large storage units constructed above the snow surface (EPA 1979). The facilities would be supported by jack-up pilings or piers resting on load-bearing plates, as shown in Figure 6.1.17. The waste canisters would be placed in cubicles inside the facility and cooled by natural draft air. The facility would be elevated above the ice surface for as long as possible to reduce snow drifting and heat dissipation. During this period, the waste canisters would be retrievable. However, when the limit of the jack-up pilings was reached, the entire facility would act as a heat source and begin to melt down through the ice sheet. It is estimated that such a facility could be maintained above the ice for a maximum of 400 years after construction (Battelle 1974).

Retrievability/Recoverability. Waste disposed of using the meltdown emplacement concept would be retrievable for a short period, but movement down into the ice and successful

deployment of the concept design would quickly render the waste essentially irretrievable. Recovery is also considered nearly impossible. Retrievability for the other two emplacement concepts is indicated in the discussions above.

6.1.5.3 Status of Technical Development and R&D Needs

Present State of Development

Ice sheet disposal is in the conceptual stage of development and an extensive R&D program would be required to implement an operational disposal system (EPA 1979 and DOE 1979). Current technology appears adequate for initial waste canister emplacement using the concepts described. Necessary transportation and logistics support systems could be made available with additional R&D. The capability of ice sheets to contain radioactive waste for long periods of time is at present only speculative, because of limited knowledge of ice sheet stability and physical properties. Verification of theories that support ice sheet disposal would require many years of extensive new data collection and evaluation.

Technological Issues to be Resolved

Key technical issues that would have to be resolved for development of the ice sheet disposal concept include:

Choice of Waste Form

- Behavior of glass or other waste forms under polar conditions
- Ability of container to withstand mechanical forces.

Design of Shipping System for Polar Seas

- Extremes of weather and environmental conditions expected
- Debarkation port design
- Ship design
- Cask design
- Recovery system for cask lost at sea.

Design of Over-Ice Transport

- Crevasse detector
- Navigational aids
- Ability to traverse surface irregularities, snow dunes, and steep ice slopes
- Maintenance of road systems
- Recovery system for lost casks.

Design of Monitoring for Emplaced Waste

- Location, integrity, and movement of emplaced canisters

- Radioactivity of water at ice-rock interface
- Hydrologic connections to open oceans and effects on ice stability.

In addition, there are serious issues connected with the ability to adequately predict long-term ice sheet behavior, including rates of motion within the sheet, the physical state and rates of ice flow, movement of meltwater at the base of the sheet, and the long-term stability of the total sheet.

R&D Requirements to Make System Operational

R&D requirements to resolve these issues may be grouped in terms of those related to the handling, transportation, and emplacement of the waste, and those related to obtaining basic information on ice sheets. In the former group, R&D would be required in the areas of waste forms (content, shape, and materials), transportation (shielding, casks, ships, aircraft, over-ice vehicles), facilities (port, handling, inspection, repair), and supply logistics (fuel, equipment, personnel requirements). Research needs applying to ice sheets would include determination of ice sheet movement and stability through geological/geophysical exploration and ice movement measurements, studies of ice flow mechanics including effects of bottom water layers, studies of global and polar climatology, and acquisition and analysis of meteorological and environmental data.

Estimated Implementation Time and R&D Costs

If the ice sheet disposal concept were to prove viable, the time required to achieve an operating system is estimated to be about 30 years after the start of the necessary research program. The research program itself would require about 15 years of activity directed primarily toward improved understanding of ice sheet conditions, selecting an emplacement method, identifying and assessing ice sheet areas most suitable for the method selected, and research and preliminary development of systems unique to the particular emplacement method and site. Should the research program culminate in a decision to proceed with project development, an additional period of 12 to 13 years would be required to implement an operational disposal system.

R&D costs for ice sheet disposal are estimated to be \$340 million (in 1978 dollars) for the initial research and preliminary development program and between \$570 million and \$800 million for development, depending on the emplacement mode chosen.

Summary

Major uncertainties, shortcomings, and advantages of the concept are summarized below:

- The environment involved is non-benign to men and equipment, and the transportation limitations are severe.
- Understanding and performance assessment of the subsurface mechanisms of transport and package degradation are not developed to any degree.
- The concept does have the capacity for multiple barriers.

- The capability for corrective action over a long period is uncertain, and site selection criteria and performance assessment capability are nonexistent.
- No site is currently, or potentially in the future, available to the U.S. for R&D.

6.1.5.4 Impacts of Construction and Operation (Preplacement)

Health impacts, both radiological and nonradiological, and natural system impacts are analyzed below.

Health Impacts

Radiological impacts would in many ways be similar to those for mined geologic disposal but would have the added problem of extensive interim storage. Nonradiologic impacts might occur both as a result of routine operations or in abnormal or accidental conditions.

Radiological Impacts. Ice sheet disposal would be different from the mined geologic repository and other alternatives because of the requirement for extensive interim storage of either processed waste or spent fuel. Such storage would be necessary because lead times for research, development, and testing are 10 to 30 years longer than those for geologic disposal (DOE 1979). During this time, radiological effects would include doses to occupational personnel, the normal release of radioactive effluents to the atmosphere, and the potential for accidental release of radioactivity. At this time, no studies are available that provide a quantitative estimate of these impacts; however, it is expected that they would be similar to those from fuel storage facilities.

Preparation of waste for ice sheet disposal would be similar to that for mined geologic disposal methods. Likewise, the radiological effects associated with this option are assumed to be similar to those associated with geologic disposal methods. The radiological risks and impacts from the transportation of the waste would be to the Arctic or Antarctic essentially the same as those discussed in seabed disposal. The ice sheet disposal option is not sufficiently developed to estimate the radiological effects of routine operations on the ice sheet.

Accidents while unloading at the ice shelf seaport or during transport over the ice could create retrieval situations that would be difficult in the polar environment. Quantitative estimates of the radiological impact of such accidents are not available.

Nonradiological Impacts to Man and Environment. Potential nonradiological impacts could occur during all phases of ice sheet disposal operations. As with many of the alternative disposal strategies, impacts can be categorized as to whether they would occur during waste preparation, transportation, or emplacement activities. In general, those impacts associated with transportation and emplacement would warrant the most analysis. Waste preparation impacts would be similar to those for other disposal strategies discussed earlier.

Occupational casualties from the nonpolar activities are expected to occur at rates typical of the industrial activities that would be involved, and to be independent of both the nuclear and polar aspects of the remainder of the system. Operations are routinely carried out with nuclear systems and in the polar regions with safety comparable to that experienced in more familiar environments. In all likelihood, the required large-scale activities could also be performed safely, with the polar conditions being reflected in higher program costs rather than in decreased safety.

Accidents in processing and handling the waste material could occur before the material reaches the embarkation facility. Impacts resulting from such accidents are common to virtually all of the alternative disposal options. Other impacts would be virtually identical to those of the subseabed disposal option because in both cases the material would be transported to a coastal location.

Nonradiological health effects for activities that would occur on the ice sheet under abnormal conditions have not been studied extensively. Occupational impacts would occur, but as stated above, it is not expected that polar conditions will significantly alter the level of effects anticipated. Non-occupational effects would be even less significant, reflecting the lack of human activity on the ice sheets.

Natural System Impacts

Quantitative estimates of the radiological impact of ice sheet disposal on the ecosystem are not available. These impacts are expected to be small because there are very few living organisms in the polar regions, except along the coastline. Nonradiological ecological impacts at the disposal site are difficult to characterize because of a lack of understanding of the processes occurring in polar environments. The present understanding of impacts on the glacial ice mass or the dry barren valleys of Antarctica is limited. The effect of the heat that would be produced by the wastes on the ice or the potential geologic host media remains unclear.

Air impacts would result from the combustion products of over-ice transport vehicles, support aircraft, and fuel consumed for heating the facilities at the various sites. At present, the effects of these products are not considered a major problem.

Few, if any, ecological impacts are expected near the disposal sites because the plant and animal life are confined mostly to the coastal areas. Access routes and air traffic lanes could be made to avoid as much as possible the feeding, nesting, and mating spots of the birds and animals that inhabit the coastal areas. Fuel spills, equipment emissions, and general transportation support activities could lead to some localized impacts along the transportation disposal corridors. Few, if any, other impacts on water are expected, except for a marginal increase in temperature of the water that would be used for once-through cooling of canisters during sea transport. The only other water uses would be for consumption by the 200 operating personnel, which would be obtained by melting the ice.

Other possible land impacts considered in the reference study include accidental spills of fuel and the probability of fuel bladders rupturing during drop-offs. Rupture of the fuel bladders is considered to be a high risk because the fuel is capable of penetrating the snow and could reach the underlying ice where it would remain until evaporated or eventually buried by additional snow. Accidental spills could reach the ocean if the incident occurred near the edge of the ice sheet.

Socioeconomic Impacts

Socioeconomic impacts for the ice sheet disposal option would be similar to those for the island and subseabed disposal options. Because these options are still at the concept level, however, detailed socioeconomic assessments are not possible. In general, socioeconomic impacts would be experienced where handling facilities are constructed and operated.

Impacts that might be expected where handling facilities would be constructed include disruptions or dislocations of residences or businesses; physical or public-access impacts on historic, cultural, and natural features; impacts on public services such as education, utilities, road systems, recreation, and health and safety; increased tax revenues in jurisdictions where facilities would be located; increased local expenditures for services and materials; and social stresses.

The operating work force required for a dock facility would likely be comparable to that for any moderate-size manufacturing facility and impacts would vary with location. Impacts would be primarily in housing, education, and transportation, with no significant impacts on municipal services. Impact costs would presumably be offset by revenues, but socioeconomic considerations at this stage are not easily quantified.

Aesthetic Impacts

Aesthetic impacts are expected to be insignificant because of the remoteness of the area and the lack of permanent residence population (EPA 1979).

Aesthetic impacts for the ocean transportation activities and embarkation facilities would be very limited and similar to those of subseabed disposal. The waste packaging and transportation activities that would be a part of the ice sheet disposal process would have aesthetic impacts similar to those of mined geologic repositories. Noise, fugitive emissions, and the appearance of facilities and equipment used to prepare and transport the waste material are common to a number of disposal options. These impacts are generally reviewed in Chapter 4.

Resource Consumption

Predisposal activities would include packaging and transportation of spent fuel to seaports for shipment to the receiving port at the ice sheet, if spent fuel were disposed of rather than reprocessed waste. If reprocessing of spent fuel were undertaken, then predisposal activities would also include conversion of the waste to a high-integrity form, like

glass, before transportation to seaports. The resource requirements of these activities have been discussed elsewhere in this document for other disposal alternatives, and would be the same for ice sheet disposal, except for differences in transportation routings.

Little quantitative information exists on the energy, resource, and land requirements unique to ice sheet disposal. Ice sheet disposal would require construction of ships, airplanes, and over-the-ice vehicles that would not be required for other disposal alternatives. A greater number of shipping casks would also be required, because of the long cask turn-around time.

Transporting the waste material to its final destination across the ice fields would also require expenditure of energy. Either surface or air transport would use large quantities of fuel because of the great distances involved.

Some land impacts would probably be experienced in connection with the embarkation port facility. An area of about 1km^2 (0.4mi^2) would be required for the shielded cell and the loading dock facilities. The port facility would be equipped with its own separate water, power, and sewer systems to assure maximum safety. The over-ice transport routes would include an area at the edge of the ice sheet, ice shelf-edge, and ice-free areas on land for unloading the shipping casks. Approximately six support and fueling stations would be required along the transport route to the disposal area. Land requirements at the disposal site are estimated at $11,000\text{km}^2$ ($4,2000\text{mi}^2$) for waste from a plant producing 5 MTHM/day based on a waste canister spacing of one/Km.

International and Domestic Legal and Institutional Considerations

The ice sheet disposal option, like the island and seabed options, would require transporting waste material over the ocean, and the general international implications of such transportation are important.

Numerous legal and institutional considerations would emerge if the ice sheet disposal concept were seriously pursued in either Greenland or Antarctica. In the case of Greenland, treaty arrangements would have to be made with Denmark because Greenland is a Danish Territory.

In the case of Antarctica, a number of treaties and agreements exist that could affect the use of the ice sheets for storage and disposal of radioactive material. Disposal of waste in Antarctica is specifically prohibited by the Antarctic Treaty of 1959, of which the United States is a signatory (Battelle 1974). The treaty may be renewed after it has been in effect for 30 years, or amended at any time.

Outcomes of two meetings reflect the current range of international attitudes toward ice sheet waste disposal. One attitude was expressed in a resolution passed by the National Academy of Sciences, Committee on Polar Research, Panel on Glaciology, at a meeting in Seattle, Washington, May, 1973. The resolution neither favored nor opposed ice sheet waste disposal as such. However, a statement from a second meeting, on September 25, 1974, in Cambridge, England, attended by scientists from Argentina, Australia, Japan, Norway, the United Kingdom, the United States, and the USSR, recommended that the Antarctic ice sheet not be used for waste disposal.

6.1.5.5 Potential Impacts over the Long Term (Postemplacement)

Potential Events

Long-term impacts with the greatest potential significance are related to glacial phenomena that are not well understood. For example, ice dynamics and climatic variations affecting glaciation might be altered by waste disposal activities. Regardless of whether meltdown, anchored emplacement, or surface storage were used, potentially major modifications in the delicately balanced glacial environment could occur.

One of the major areas of uncertainty stems from our limited understanding of ice sheet conditions. Little is known of the motion of the continental ice sheets except for surface measurements made close to the coast (Gow et al. 1968). Three general types of flow have been defined--sheet flow, stream flow, and ice-shelf movement (Mellor 1959). Each type of flow appears to possess a characteristic velocity. It is also believed that ice sheets where bottom melting conditions exist may move almost as a rigid block, by sliding over the bedrock. Where there is no water at the ice-bedrock interface, it is believed that the ice sheet moves by shear displacement in a relative thin basal layer. The formation of large bodies of water from the waste heat could affect the equilibrium of such ice sheets.

In addition, two potential problems concerning the movement of the waste are unique to an ice sheet repository. First, the waste container would probably be crushed and breached once it reached the ice/ground interface as a result of ice/ground interaction. Second, the waste might be transported to the sea by ice movement.

Compared with other disposal schemes, the probability of human intrusion would be very low because the disposal area would be located in the most remote and inaccessible part of the world, presently with a low priority for exploration of natural resources or habitation. The lack of human activity in these areas would markedly decrease the chance of humans disturbing waste material emplaced in an ice sheet. Conversely, because of the remoteness of these areas they are relatively unexplored. Therefore they could attract considerable future resource exploration.

Potential Impacts

After the waste is emplaced and man's control is relinquished or lost, possible impacts fall into two broad categories. One of these relates to the reappearance of the radioactive waste in the environment, and the other involves the chance that the presence of waste would trigger changes in the ice sheets that would have worldwide consequences. For options that would place the waste within the ice or at the ice/ground interface, significant research would be required to predict future ice movements, accumulation or depletion rates, subsurface water flow rates, frictional effects at the interface, and trigger mechanisms. A major purpose of this research would be to compare the degree of sensitivity of the predicted behavior to man's ability to forecast long-term situations such as global weather patterns, stability of the ice sheets, and sea-level changes.

Specific areas of concern, as discussed below, are:

- Effects of waste on ice sheet environment
- Effects of ice sheet on waste
- Effects of waste on land environment.

Effects of Waste on Ice Sheet Environment. If waste canisters were allowed to reach or approach the bottom of the ice, they could possibly generate sufficient heat to produce a water layer over a large portion of the bottom surface of the ice. Furthermore, melt pools around the canisters could conceivably coalesce and also unite with any subglacial water, in the disposal area, to form a large water mass within the ice or at the edge of the ice-bedrock interface. Either event might trigger an increase in the velocity of the ice mass and perhaps produce surging. It has been postulated that major surges in the East Antarctica ice sheet could affect solar reflection and alter the sea level. The most extreme effect would be the start of glaciation in the Northern Hemisphere (Wilson 1964). The accelerated movement could also move emplaced material toward the edge of the ice sheet, possibly reducing the residence time. Basal ice sheet water could also conceivably form a pathway for transporting waste material from the disposal area to the edge of the ice sheet, and thus to the ocean.

Hypothetical dose calculations have been made for radionuclides released from an ice sheet disposal site into the ocean off the coast of Greenland (EPA 1979). On the basis of assumptions that a failure occurs in the disposal system, the release of radionuclides into the Greenland current of 8×10^6 m³/sec would be 0.3 percent/yr of the total inventory available. Complete mixing could occur in the ocean. Human pathways are assumed to be mostly via fish consumption. The maximum dose was considered to be from an individual consuming 100 kg/yr of fish caught in these contaminated waters and is estimated to be 0.2 mrem/yr. Further discussion of radioactive releases to the ocean is included in Section 6.1.4.5 on the subseabed concept.

Effects of Ice Sheet on Waste. Movement of the ice sheet might cause shearing or crushing of canisters, allowing water to come in contact with the waste form so that leaching could occur. Such breakage would most likely occur when the canisters are moved along the ice-bedrock interface.

If major climatic changes were to produce an increase in temperature in the polar region, the ice sheet might erode to such an extent that it would allow the waste to be much closer to the edge of the ice. The temperature increase could also increase the velocity of the ice movement toward the coast.

Effects of Waste on Land Environment. As in the case of space and subseabed disposal, geologic repository facilities are assumed to be constructed for TRU and other wastes not disposed of through the procedures established for the majority of HLW. Long-term effects could result from these auxiliary activities. These impacts would be similar to those

described for the mined geologic concept. The other land area that could be impacted is the region of dry barren valleys in Antarctica. If wastes were placed in this area, impacts would be very similar to those of the mined geologic repository. The major difference would be that the ground-water regime in Antarctica would mostly affect remote frozen ground-water systems.

Terrestrial ecosystems in the ice sheet regions under study for disposal sites are limited in diversity. Severe climatic conditions limit most organisms to the seaward margins of both Greenland and Antarctica. Consequently, the potential for impact to terrestrial organisms in the ice sheet disposal is quite limited. Potentially more significant are the long-term ecological effects of any accidents that would occur on the land mass where the wastes were generated. As described in Section 5.6, these impacts should not be significant unless an accident or encroachment occurs.

6.1.5.6 Cost Analysis

The cost of depositing nuclear wastes in ice sheets is currently expected to be relatively high; higher, for example, than the cost of geologic emplacement in the U.S. This is primarily because of the high costs for R&D as presented in Section 6.1.5.3. Capital, operating, and decommissioning cost estimates are presented below.

Projected Capital Costs

Projected capital costs for ice sheet emplacement of 3000 MT/yr of spent fuel, or the wastes recovered from processing that amount of fuel, are \$1.4 billion to \$2.3 billion as shown in Table 6.1.17.

Projected Operating Costs

Projected operating costs for the emplacement of 3000 MT/yr of spent fuel or HLW are shown in Table 6.1.18.

Decommissioning Costs

Decommissioning costs associated with contaminated equipment would probably be limited primarily to the shipping casks used to transport waste canisters for ice sheet disposal. These costs are estimated at \$9.7 million, which is 10 percent of the initial capital cost of the shipping casks. Costs for decommissioning other facilities and equipment are assumed to be similar to those for other waste disposal alternatives.

6.1.5.7 Safeguard Requirements

Because the reference concept uses both ice sheet and mined geologic disposal, its implementation would require safeguarding two separate disposal paths. The risk of diversion for the meltdown concept would be basically a short-term concern because once the waste had been successfully disposed of in accordance with design, it would be considered irretrievable. For the anchored and surface storage concepts, although the waste would be considered retrievable for as long as 400 years, the harsh environment in which it would be

TABLE 6.1.17. Capital Costs For Ice Sheet Disposal
(Millions of 1978 Dollars)

Case I. Meltdown or Anchored Emplacement: Surface Transportation

1. Construction of Port Facilities	730
2. Sea Transport Vessels	290
3. Ice Breakers	190
4. Over-Ice Transport Vehicles	100
5. Drilling Rigs	50
6. Monitoring Equipment	50
7. Shipping Casks	100
8. Aircraft	100
9. Support Facilities	<u>150</u>
	1760

Case II. Surface Storage

1. Construction of Port Facilities	730
2. Sea Transport Vessels	290
3. Ice Breakers	190
4. Over-Ice Transport Vehicles	100
5. Surface Storage Facility	500
6. Monitoring Equipment	50
7. Shipping Cask	100
8. Aircraft	100
9. Support Facilities	<u>190</u>
	2250

Case III. Meltdown or Anchored Emplacement: Aerial Emplacement

1. Construction of Port Facilities	500
2. Sea Transport Vessels	150
3. Aircraft	500
4. Shipping Casks	100
5. Monitoring Equipment	50
6. Support Facilities	<u>150</u>
	1450

TABLE 6.1.18. Operating Costs For Ice Sheet Disposal
(Millions of 1978 Dollars/Year)

Emplacement Concept Emplacement Method	<u>Meltdown or Anchored</u>		<u>Surface Storage</u>
	<u>Surface</u>	<u>Aerial</u>	<u>Surface</u>
Cost Category:			
Operating Personnel(a)	34	29	39
Material & Consumables(b)	58	29	58
Services & Overhead(c)	68	58	78
Capital Recovery(d)	<u>175</u>	<u>141</u>	<u>224</u>
Total	335	257	399

(a) Based on \$50,000/man-year.

(b) Including \$29 million/yr and \$5 million/yr port upkeep for surface and aerial emplacement, respectively.

(c) Based on twice the operating personnel costs.

(d) Based on 10 percent of capital expenditures (not including research and development costs). Encapsulation costs not included.

placed and the equipment needed for retrieval would also make any risk of diversion primarily a short-term concern. Only minimum safeguards would be required after emplacement. Physical protection of the sensitive facilities and transportation operations would be the most effective way to deny access to the waste for the short term, as is common to most waste disposal alternatives. See Section 4.10 for additional discussion of predisposal operation safeguard requirements.

6.1.6 Well Injection

6.1.6.1 Concept Summary

Well injection technology was initially developed by the oil industry for the disposal of oil field brines. These brines were usually pumped back into the original reservoir and, in some cases, used to "drive" the oil toward a producing well. The well injection concept has subsequently been used for the disposal of various natural and industrial wastes. The techniques developed in the oil industry handle liquid wastes only - particulate matter can cause blocking of the pores in rock.

A well injection process using grout was developed by Oak Ridge National Laboratory (ORNL) for the injection of remotely handled TRU liquid radioactive wastes into shale strata (ERDA 1977). This technique is also suitable for grout slurry wastes, and a new facility is now under construction at ORNL for liquid and slurry waste injection (ERDA 1977). Well injection could be a low cost alternative to deploy and operate because of the widespread use of the required techniques and the "off-the-shelf" availability of the main components. Two reference methods of well injection are considered in this section: deep well liquid injection and shale grout injection.

Deep well injection would involve pumping acidic liquid waste to depths of 1,000 to 5,000 m (3,300 to 16,000 ft) into porous or fractured strata suitably isolated from the biosphere by overlying strata that are relatively impermeable. The waste may remain in liquid form and might progressively disperse and diffuse throughout the host rock. This mobility within the porous host media formation might be of concern regarding release to the biosphere. Questions have also arisen regarding the possibility of subsequent reconcentration of certain radioisotopes because of their mobility. This could lead to the remote possibility of criticality if, for instance, the plutonium is reconcentrated sufficiently. Isolation from the biosphere would be achieved by negligible ground-water movement in the disposal formation, particularly towards the surface, retention of nuclides due to sorption onto the host rock mineral skeleton, and low probability of breaching by natural or man-made events. The concept is not amenable to a multiplicity of engineered barriers.

For shale grout injection, the shale would first be fractured by high-pressure water injection and then the waste, mixed with cement and clays, would be injected into suitable shale formations at depths of 300 to 500 m (1,000 to 1,600 ft) and allowed to solidify in place in layers of thin solid disks. The shale has very low permeability and probably good sorption properties. The injection formations selected would be those in which it could be shown that fractures would be created parallel to the bedding planes and would therefore remain within the host shale bed. This requirement is expected to limit the injection depths to the range stated above. Direct operating experience is available at ORNL for disposal of TRU wastes by shale grout injection. The grout mixes have been designed to be leach resistant and hence the concept minimizes the mobility of the incorporated radioactive wastes.

Isolation from the biosphere is achieved by low leach rates of radionuclides from the hardened grout sheet, negligible ground-water flow particularly up through the shale strata, retardation of nuclide movement by minerals within the shale strata, and low probability of breaching by natural or man-made events.

6.1.6.2 System and Facility Description

System Options

The two reference concepts for well injection disposal of nuclear waste have been selected from a number of options available at each step from the reactor to disposal at the well injection facility. These two concepts are judged as "most likely" based on the status of current technology. A summary of various options to be considered for well injection disposal is illustrated in Figure 6.1.18. Additional pertinent data available on the options can be found in various source material listed in Appendix M.

Waste-Type Compatibility

For both reference concepts the waste form injected would be HLW. Since disassembly and some processing would be necessary for well injection, the concepts would be suitable for fuel cycles that recycle uranium and plutonium. However, well injection could also be applied to once-through fuel cycles after dissolution or slurring of spent fuels. In these

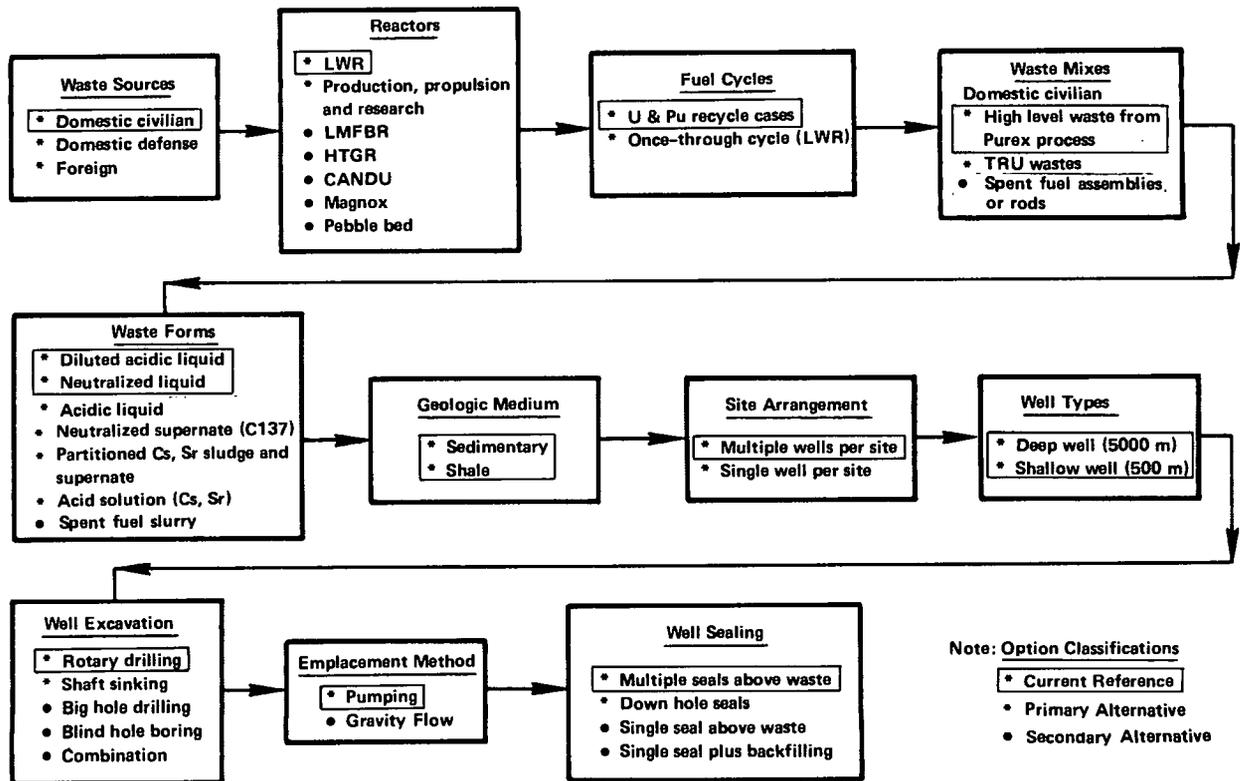


FIGURE 6.1.18. Major Options for Well Injection Disposal of Nuclear Waste

cases, the injection liquid would contain large amounts of actinides, which might affect the thermal properties and interaction mechanisms of the waste in the host media. Well injection might also be used to dispose of high-heat-level partitioned wastes, which could relieve high thermal loadings in a mined geologic repository for example. Note that retrieval would be difficult and incomplete using either concept, although deep well injection would have more potential for at least partial retrieval than would the shale-grout method, which would fix the waste in a relatively insoluble solid.

For deep well injection, the liquid waste would have to be substantially free from all solid matter to prevent clogging of the formation pores. Filtration down to 0.5 μ m particles is typical for process waste injection systems (Hartman 1968). The waste would have to remain acidic to ensure that all the waste products stay in solution.

For shale grout injection, neutralized waste (sludge and supernate) would be mixed with cement, clay, and other additives.

Waste System Description

The fuel cycle and process flows associated with the two reference options are illustrated on Figure 6.1.19. Significant features of these concepts are summarized in Table 6.1.19.

Both concepts are based on restricting the maximum temperature in the injection formation to 100 C (212 F), assuming a geothermal gradient of 15 C/km (44 F/mile), to avoid undesirable mineralogical effects that would occur at higher temperatures. (For example, comparatively large amounts of waste would be released from the clay mineral montmorillonite if

TABLE 6.1.19. Reference Concepts Summary (DOE 1979)

Reference Concepts	Depth of Injection	Disposal Formation
Deep well liquid injection	100-m-thick zone at average depth of 1,000 m	Sandstone with shale caprock at 950-m depth; porosity 10 percent
Shale grout injection	100-m-thick zone at average depth of 500 m	Shale extending to within 50 m of ground surface

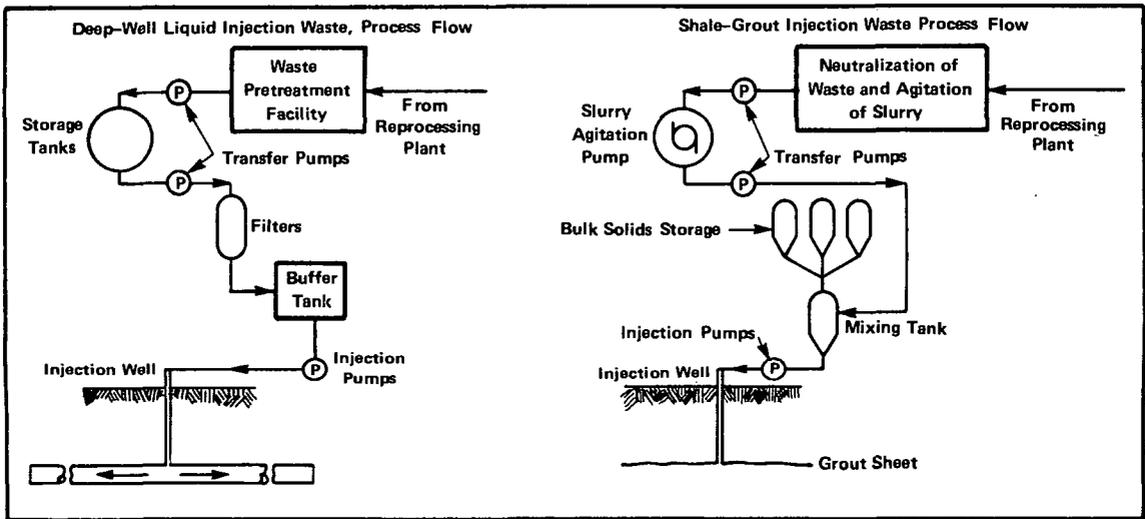
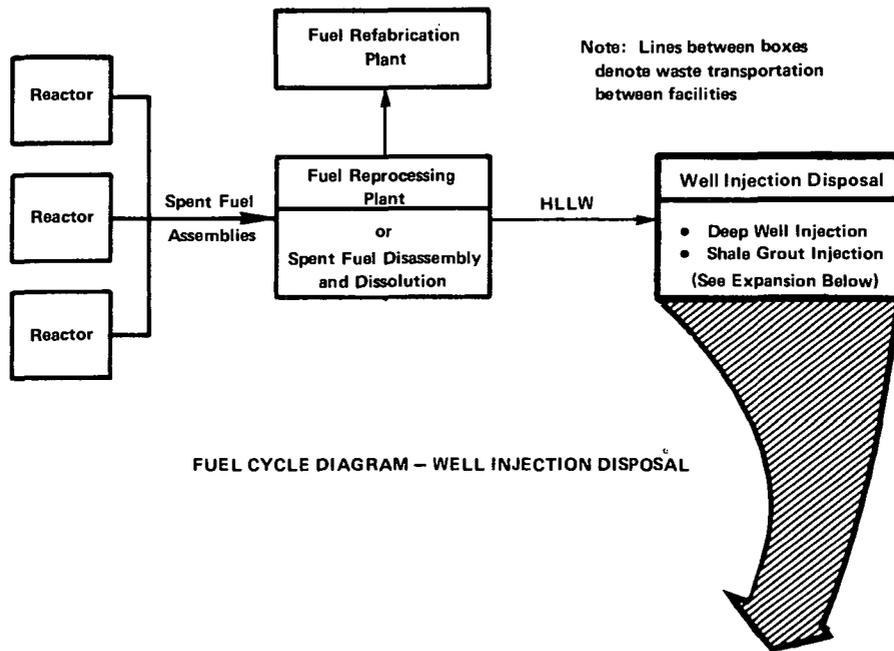


FIGURE 6.1.19. Waste Management System--Well Injection Disposal

heated to above 100 C) (EPA 1973). Although disposal strata containing more inert minerals, particularly quartz-rich sandstones suitable for deep well liquid injection, might sustain higher temperatures, thermal effects on containment formations, which may include temperature sensitive minerals, would also have to be considered.

Deep Well Injection

In the deep well injection concept, the liquid wastes would be fed into porous or fractured strata, such as depleted hydrocarbon reservoirs, natural porous strata, or zones of natural or induced fractures. To protect freshwater aquifers from waste contamination, the injection zone would have to be well below the aquifers and isolated by relatively impermeable strata, e.g., shales or salt deposits.

In general, injection requires pressure at the wellhead, although in some circumstances gravity feed is sufficient. The controlling factors are the rate of injection and the permeability of the disposal formation. The increase in the total fluid volume in an injection zone is accommodated by compression of any fluid already present and expansion of the rock formation. The relation between injection rates and pressures is based on extensive oil-well and ground-water experience. Injection is possible at depths down to several thousand meters.

For this concept, the activity of the injection waste has been assumed to be controlled by the allowable gross thermal loading, the injection zone thickness, and the porosity in that zone. It is also assumed that one injection zone with two wells would be used at each site. In the long term, the waste might progressively disperse and diffuse throughout the host rock and eventually encompass a large volume. The concentration might be variable and unpredictable. Thus, criteria for permissible activity levels might be required. Determination of the dilution requirement is complicated by the sorption of nuclides onto the mineral skeleton, to an extent determined by waste chemistry and rock mineral content. If sorption were too high, concentration of heat-generating components might result in "hot spots".

Injected waste might be partially retrieved by drilling and pumping, but sorption of nuclides onto the mineral skeleton and precipitation within the pores would limit the amounts recovered.

Predisposal Treatment. In deep well injection, spent fuel would be shipped to a processing facility at the well injection site. The spent fuel would be dissolved in acid and the hulls removed. (For recycle, the uranium and plutonium would be removed from the acid solution.) The acid solution would constitute the basic waste form for isolation.

The acid waste from reprocessing would contain both fission products and actinides. Between 60 and 75 percent of the heat generated in the initial emplacement years would be due to ^{90}Sr and ^{137}Cs . Partitioning strontium and cesium from the remainder of the waste

would permit different isolation practices to be adopted for the high-heat-generating, relatively short-lived isotopes (half-lives about 30 years) and the remainder of the waste containing the much longer lived, lower heat generating isotopes.

The liquid waste would be diluted with water or chemically neutralized and pumped from the reprocessing facility to the injection facility or to interim storage in holding tanks.

Site. Deep well injection would require natural, intergranular fracture porosity or solution porosity formations, overlain by impermeable cap rock, such as shale. A minimum acceptable depth for disposal would be about 1,000 m (3,300 ft) (EPA 1973). The injection site must not conflict with either present or future resource development.

Synclinal basins would be particularly favorable sites for deep well liquid injection since they consist of relatively thick sequences of sedimentary rocks frequently containing saline ground water (Warner 1968). Ground-water movement within the injection formation would have to be limited, however, particularly vertical movement.

The lithological and geochemical properties of the isolation formation would have to be stable so that the behavior of the waste could be accurately predicted. In general, sandstone would be the most suitable rock type because it combines an acceptable porosity and permeability with chemically inert characteristics relative to the acid waste form.

The overall site area has not been determined yet, but would be greater than the 1270 ha (3140 acres) initial injection area and would depend on the maximum horizontal dimension of the injection area, the size of control zone required around the repository, and the total amount and type of waste to be injected.

Drilling System. The drilling rigs would be similar to those used in the gas and petroleum industries and would be portable for movement from one location to another on the site. Each complete rig would require a clear, relatively flat area, approximately 120 m x 120 m (400 ft x 400 ft) at each hole location (see Section 6.1.1).

Repository Facilities. The processing plant would be located on site as an integral part of the overall injection system. The basic repository facilities would be similar to those required for the very deep hole concept, as discussed in Section 6.1.1 (Bechtel 1979a).

Interim storage tanks similar to those described for the rock melt concept (Section 6.1.2) would be provided for surge capacity. The stainless steel tanks would have a combined capacity of about 10^6 liters (2.8×10^5 gal) which equals 3 months production. The tanks would be similar in design to those at the AGNS plant in Barnwell, South Carolina, which are contained in underground concrete vaults and provided with internal cooling coils and heat exchangers to prevent the waste from boiling.

An underground pipeway system would connect the reprocessing facility to the storage tanks and the injection facility. The pipe would be double cased and protected by a concrete shielding tunnel with leak detectors provided in the annulus of the pipe. The pipeway design would provide containment, monitoring, decontamination, maintenance, and decommissioning

capabilities, primarily performed remotely. A heavy concrete and steel confinement building would provide containment for the well and injection operations and shielding for the radioactive systems.

Sealing Systems. The well hole would probably be sealed by a combination of borehole seals and backfilling, using a procedure similar to the one discussed for the very deep hole concept (Section 6.1.1).

Retrievability/Recovery. Liquid waste that had been injected might be partially retrievable by conventional well techniques. Although much of the waste might be physically or chemically sorbed by host geologic media, some species, in particular, ^{137}Cs , would be expected to remain in at least partially retrievable solution.

Shale Grout Injection

In the shale grout injection process, neutralized liquid waste or an irradiated fuel slurry would be mixed with a solids blend of cement, clay, and other additives, and the resulting grout would be injected into impermeable shale formations. The initial fracture in the shale would be generated by hydrofracturing with a small volume of water. The injection of waste grout into this initial fracture would generate sufficient pressure to propagate a thin horizontal crack in the shale. As injection of the grout continued, the crack would extend further to form a thin, approximately horizontal, grout sheet, several hundred feet across. A few hours after injection, the grout would set, thereby fixing the radioactive wastes in the shale formation. Subsequent injection would form sheets parallel to and a few feet above the first sheet.

The principal requirement for shale grout injection is that the hydrofracture, and hence the grout sheet, develops and propagates horizontally. Vertical or inclined hydrofractures could result in the waste gaining access to geologic strata near the surface, and even breaking out of grout at the bedrock surface itself. Theoretical analyses indicate that, in a homogeneous isotropic medium, the plane of hydrofracture develops perpendicularly to the minor principal stress (NAS 1966). Thus, a requirement for horizontal hydrofracturing is that the horizontal stresses exceed the vertical stresses.

On the basis of work at ORNL, approximately 40 injection wells would be required at each of five facilities. The activity level for the shale grout injection alternative is based on the reference concept (Schneider and Platt 1974) of 40 Ci/l activity in the initial grout. The acceptable gross thermal loading (GTL) could be assured by controlling the number of grout injections in the disposal formation. Depending on the fuel cycle, the maximum number of 2-mm (0.08-in.)-thick grout layers would be five to seven per injection site.

Site. A thick sequence of essentially flat-lying shale strata would be required for shale grout disposal, with in situ stress conditions favorable for the propagation of horizontal hydrofractures. Such conditions are generally found to a maximum depth of 500 to 1,000 m (1,650 to 3,300 ft). As with deep well liquid injection, the site would have to be located to preclude conflicts with resource development.

Shale deposits in the United States have been studied for suitability for underground waste emplacement (Merewether et al. 1973). The studies conclude that shale, mudstone, and claystone of marine origin in areas of little structural deformation, low seismic risk, and limited drilling are generally most promising. These include the Ohio shale of Devonian age in northern Ohio and the Devonian-Mississippian Ellsworth shale and the Mississippian-coldwater shale in Michigan. In the Rocky Mountain states, the Pierre shale and other thick shales of late Cretaceous age are also potential host rocks.

The overall site area for shale grout injection has not been determined yet, but it would be greater than the 1270 ha (3140 acres) initial injection area and would depend on the maximum horizontal dimension of the injection area and the size of the control zone required around the repository.

Drilling System. The drilling system for shale grout injection would be similar to that for deep well injection.

Repository Facilities. Repository facilities for shale grout injection would be identical to those for deep well injection with the exception of additional high-pressure pumps for fracturing and equipment related to mixing the grout with the liquid waste prior to injection (see Figure 6.1.19).

Sealing Systems. The repositories would be sealed in the same manner as deep well holes.

Retrievability/Recovery. Wastes disposed of by this concept would be essentially irretrievable because of the fast solidification and stability of the waste-grout mixture. Total recovery of the wastes would likely involve extremely difficult and extensive mining operations to excavate the rocklike waste form.

6.1.6.3 Status of Technical Development and R&D Needs

Present State of Development and Technological Issues

The basic techniques required for well injection of fluids and grouts have been developed in the course of many projects undertaken by the oil and chemical industries for the disposal of nonradioactive toxic and nontoxic wastes. In addition, limited disposal of radioactive waste grouts has been successfully completed at ORNL (ERDA 1977, Delaguna et al. 1968).

Geology. The geology of sedimentary basins in the United States has been examined extensively with a view to suitability for deep well liquid injection of radioactive wastes, and reports are available covering several areas.^(a) In addition to these studies, a large

(a) See Repenning 1962, Sandberg 1962, Beikman 1962, MacLachlan 1964, Legrand 1962, Repenning 1959, Colton 1961, and DeWitt 1961.

volume of geologic data (stratigraphy, lithology, petrography) exists for potential disposal areas. These data have been gathered for basic geologic research or as a result of resource exploration and exploitation. However, the existing data are considered suitable for only conceptual, generic studies and identification of candidate sites.

Geohydrology. Modeling to predict waste extent and nuclide transport would be required for both liquid and grout injection. In the past decade, numerical modeling methods using finite-difference and finite-element techniques have been developed using available high-speed digital computers (Pinder and Gray 1977, Remson et al. 1971). Two- and three-dimensional fluid-flow techniques with thermal and stress dependency are available. Computer codes also exist for the analysis of radionuclide transport, including the effects of decay, adsorption, and dispersion (Burkholder 1976). However, these analytical techniques are limited because of an insufficient data base and incompletely defined constitutive parametric relationships.

State-of-the-art testing techniques include the use of multiple devices to isolate sections of the borehole. These devices provide for reduction in measurement error through improved control of bypass leakage. The multiple devices also help determine directional permeability (Maini et al. 1972). Multiple hole analyses are used to define the direction and magnitude and measure of rock mass permeability (Rocha and Franciss 1977, Lindstrom and Stille 1978). Because rock properties are directionally dependent, particular consideration must be given to methods of analyzing field data before a well injection site could be chosen.

Drilling and Injection Technology. The well injection disposal would require relatively simple engineering design, construction, and operation. Oil well drilling technology, fundamental to the concept, is available and well proven.

The deep well injection disposal method has been applied in the United States for natural wastes, in particular, oil-field brines, and for industrial wastes, such as steel pickle liquors, uranium mill wastes, and refinery and chemical process wastes^(a). The deepest waste injection well completed and operated to date was at Rocky Mountain Arsenal, where fractured Precambrian gneiss, at a depth of 3,660 m (12,000 ft), was used as the disposal formation (Pickett 1968).

Shale grout injections of remotely handled TRU wastes have been carried out at ORNL at a depth of about 275 m (900 ft) (ERDA 1977). Over 6.8×10^6 l (1.8×10^6 gal) of waste containing primarily ^{137}Cs (523,377 Ci) with a lesser amount of ^{90}Sr (36,766 Ci), together with minor quantities of other radionuclides have been injected over 10 years.

(a) Such applications are described in DeWitt 1961, Pinder and Gray 1977, Remson et al. 1971, Burkholder 1976, Maini et al. 1972, Rocha and Franciss 1977, Trevorrow et al. 1977, Lindstrom and Stille 1978, White 1965, Hult et al. n. d., Pickett 1968, Warner and Orcutt 1973, Lunn and Arlin 1962, Clebsch and Baltz 1967, Spitsyn et al. 1973, Capitant et al. 1967, and Roedder 1959.

Waste Preparation Technology. Liquid waste might require pretreatment to ensure compatibility with the rock. No operating injection facilities exist at present for high-level acid wastes. Pretreatment for most industrial wastes comprises filtration and limited chemical treatment. Since well injection is usually being pursued to reduce waste processing requirements, chemical treatment is minimal, and may include the addition of biocides and chloride to prevent plugging of the well from bacterial growth (Hartman 1968).

Waste preparation for shale-grout injection at ORNL has been the subject of extensive testing to develop an economical mix with good pumping and leach-rate characteristics (Moore et al. 1975, Hollister and Weimer 1968). Research indicates that the use of ash as a partial substitute for cement reduces costs and enhances strontium retention. Mixes incorporating various clays and grout shale have been tested. Leach rates of 3.2×10^{-5} g/cm²/day for strontium and 2.1×10^{-6} g/cm²/day for cesium have been obtained. The latter value is approximately equivalent to the leach rate for borosilicate glass (ERDA 1977).

Isolation and Safety. Isolation and safety analyses are based on

- Definition of source term (concentration, form, location, time)
- Characterization of pathway (transport velocity, chemical or physical changes, path length barriers, ecosystems involved)
- Exposure and "dose-to-man" calculations for both specific groups and total population.

A range of data values for the parameters can be analyzed to provide a probabilistic basis for the results. Methods involving modeling and analysis of failure processes have been employed for analyzing the performance of conventional disposal options (Logan and Berbano 1977) and would also be applicable to deep well injection concepts.

R&D Requirements

Since experience in the basic techniques required for well injection exists, the uncertainties associated with the design basis are related primarily to extrapolation of this experience to other waste forms, to other geologic settings, and to modified quantities and disposal rates. There are already techniques for preparing radioactive wastes in liquid or slurry form; however, there are uncertainties in formulating liquid wastes that would provide stability and compatibility with the disposal formation. For slurries, further R&D would be required for the development of optimum mixes, which would be related to the specific characteristics of the waste and disposal formation.

Geologic formations suitable for the injection of waste would have to be identified and verified on a site-specific basis. The exploratory techniques needed to do this are in an early stage of development, and would require further R&D with particular emphasis on verifying local geologic structure, establishing local and regional geohydrologic conditions, determining thermal and mechanical properties and in situ stresses, and locating and orienting discontinuities.

With the basic technology for injecting radioactive wastes into geologic strata already available, these research and development requirements can be categorized into several discrete areas of development, as described below.

System Data Base. It would be essential that the total R&D program be supported by a data base that covered all the components that could affect performance of the disposal system. The data base would cover the waste form, its modification, storage and injection, and the characteristics of the disposal formation from near to far field.

Development of Criteria for and Categorization of Siting Opportunities. The two types of well injection disposal methods, liquid and grout injection, would require significantly different but clearly definable disposal formation characteristics. Disposal site selection would have to proceed in stages, starting with the derivation and assembly of specific criteria, followed by successive narrowing of the field of choice to a specific site or sites. This approach would provide valuable generic hydrogeological data at an early stage for subsequent use in other R&D studies. The selection process could be undertaken initially using available geologic and hydrologic data and techniques. At the site-specific level, however, the use of yet-to-be developed "nonpenetrative" techniques might be required to minimize the amount of down-hole exploration.

Liquid and Slurry Wastes. A key facet of well injection is pretreatment of the liquid or slurry to a form that would be both compatible with the receiving formation and also the best use of the potential of that formation to fix and retain the nuclides. Optimum forms and requisite admixtures would have to be identified. The R&D program would have to proceed from the generic to the specific when the geochemistry of the disposal formation is known.

Techniques for Predicting the Configuration of Injected Wastes. Fundamental to the concept of "safe" disposal of waste is the necessity to predict, with a high degree of accuracy, the configuration that the injected wastes, whether liquid or grout-fixed slurry, would adopt in the disposal formation for both the short and long term. The technology should provide this capability.

For the liquid injection method, predictive capability is currently limited by the existing data base. Numerical simulation techniques are available, but these do not cover the range of conditions that might be encountered. Mathematical models for geohydrological and geochemical interaction studies would be needed.

"Nonpenetrative" Exploration Techniques. The presence of a drill hole could impair the isolation of a disposal site. At present, the majority of exploratory techniques require drilling at least one hole (and often several) to obtain reliable information from geological strata. R&D would be needed to develop nonpenetrative exploration techniques, similar to other geologic disposal methods.

Sealing Systems. It is assumed that the sealing system for well injection would have to meet the same time requirements for sealing penetrations that a mined repository must meet. The primary purpose of the seal is to inhibit water transport of radionuclides from the waste

to shallow ground water or to the surface for an extended time period. Expansive concretes make the best seals under current technology and do so at an acceptable cost. However, current experience with seals, whether of cement, chemical, or of other materials, is only a few years old. Further development of sealing technology would, therefore, be required (Bechtel 1979a). For integrity to be maintained, the sealing material would have to meet the following requirements:

- Chemical composition - the material must not deteriorate with time or temperature when compared to host rock characterization.
- Strength and stress-strain properties - the seal must be compatible with the surrounding material, either rock or casing.
- Volumetric behavior - volume changes with changes in temperature must be compatible with those of enclosing medium.

The sealing system for well injection would consist not only of plugs within the casing, but also of material to bridge the gap between casing and competent rock not damaged by drilling. To minimize possible breaks in containment, rigorous quality assurance would be required during emplacement of several high quality seals at strategic locations within the borehole.

Research and development would be needed in two major areas - material development and emplacement methodology - to ensure complete isolation. Material development would include investigating plugging materials (including special cements), compatible casing materials, and drilling fluids. Because the seal would include the host rock, these investigations should include matching of plugging materials with the possible rock types. It is conceivable that different materials would be required at different levels in the same hole.

Emplacement methodology would have to be developed for the environment of the hole. Considerations would include operation in the aqueous environment, casing and/or drilling, and fluid removal. Because the emplacement methodology would depend on the type of material, initial studies of material development would have to precede emplacement methodology development. However, the two investigations would be closely related and would interface closely. In situ tests would have to be performed to evaluate plugging materials. Equipment developed would include quality control and quality assurance instrumentation.

Monitoring Techniques. In common with other methods of underground disposal, techniques would be required for monitoring the movement/migration of radioactive material from the point of emplacement.

Borehole Plugging Techniques. Borehole plugging techniques would require development at an early stage to permit safe exploration of candidate sites.

Implementation Time and Estimated R&D Costs

The R&D program described above is generic. Specific estimates for required implementation time and R&D costs would depend on the details of the actual development plan, and are deferred pending plan definition.

Summary

Major uncertainties, shortcomings, and advantages of the concept are summarized below:

- The concept is not compatible with the multi-barrier philosophy, relying only on a potentially non-inert waste form and the geology.
- Performance assessment and siting technology for HLW injection are essentially non-existent.
- Retrieval, technical conservatism, and adequate design margins do not appear possible due to the diffuse nature of the emplaced material.
- The emplacement technology is considered to be essentially available.

6.1.6.4 Impacts of Construction and Operation (Preemplacement)

In some respects the environmental impacts of the well injection concepts are better understood than the impacts from the other disposal alternatives. This is because of their current use--deep well by the oil and gas industry to dispose of chemical waste and shale grout injection by the Oak Ridge National Laboratory to dispose of remotely handled TRU wastes. Potential use of well injection for disposing of long-lived or high-level radioactive waste, however, has not been demonstrated.

Although quantitative estimates of environmental impacts of well injection have not been made, it is expected that many of the impacts would be essentially the same for the two reference concepts.

Health Impacts

Radiological Impacts. The radiological impacts from routine operations during most phases of well injection disposal (e.g., reactor spent fuel storage, and intermediate spent fuel storage) are expected to be the same as those for a mined geologic repository. However, the extra operation to reprocess spent fuel from the once-through fuel cycle to produce a liquid solution or grout could be expected to add to the radiological impacts. Quantitative estimates of these impacts are not available at this time. Likewise, the radiological impacts associated with the transportation of wastes are expected to be similar to those for a mined geologic repository, with the exception of transporting HLW from the reprocessing plant. Since, for the reference repositories, the injection facility is adjacent to the reprocessing plant, the need to transport HLW is eliminated, which thereby reduces the corresponding radiological impact.

Unavoidable environmental effects of the well injection option would include operational radiation doses to facility workers involved in injection or maintenance and repair. Design and operational procedures would be directed to reducing doses to the lowest levels possible. At the ORNL remotely handled TRU waste facility the radiation exposure per man per grout injection has averaged 0.025 rem during injection operations and 0.188 rem during preinjection maintenance (ERDA 1977). However, the data are not sufficient to determine whether these occupational exposures would be applicable to an HLW repository. Accident scenarios

may be conveniently divided into surface and subsurface events. Surface operating accidents would include pipe ruptures and spills, failure of transfer or injection pumps, and loss of necessary cooling to the storage tanks. To minimize risk, normal nuclear engineering design strategies would be required, with redundancies incorporated into all critical systems and components (for example, pumps, power supply, and monitoring equipment). Subsurface accidents, for which contingency plans would have to be prepared, would include well-pipe rupture, equipment failures, uncontrolled fracture development (shale grout injection), and penetration of waste through the containment formation due to highly permeable features, abandoned or poorly sealed wells, or exploration or monitoring of drill holes. Site exploration and analyses would be directed toward minimizing the probability and the effects of subsurface failures.

Presently, there are no quantitative estimates of the radiological impacts of such accidents to occupational personnel, nonoccupational personnel, or the ecosystem. Furthermore, since the waste would be in a nonsolid form for well injection, the radiological impacts are not expected to be similar to those resulting from accidents at a mined geologic repository.

Nonradiological Impacts. Little formal study has been completed on the nonradiological health effects of the well injection disposal process. In general, predisposal activities, such as fuel handling, storage, transportation, and reprocessing, for both reference concepts would be the same as for a mined geologic repository. Pretreatment of the disposal formation with acid, however, might be required. Although potential impacts have not been quantitatively assessed, it can be concluded that nonradiological health effects would result from handling this hazardous material.

Because wastes injected into the wells would have to be in liquid or grout form, two important differences are anticipated between well injection and mined geologic disposal. First, the well injection disposal site would have to be at the same place as the reprocessing facility. Colocating these facilities would minimize the transportation requirements and associated risks. It would also reduce some of the nonradiological impacts associated with transportation activities.

Second, well injection would involve surface and subterranean activities with different hazards than those associated with mined geologic disposal--formation drilling and fracturing, compared to large-scale excavation, are the principal below-ground activities that could lead to nonradiological health impacts. Preparing the wastes for disposal would involve facilities designed to mix the wastes with clay, cement, and other additives for the shale-grout method. For the liquid injection process, more limited mixing facilities would be needed. In either case, studies completed to date have not identified significant nonradiological impacts for these activities under routine operating conditions. Under abnormal conditions, pipe ruptures and spills, failure of injection pumps, and other problems discussed under radiological impacts could lead to nonradiological impacts as well.

Natural System Impacts

Effects on the ecosystem near a well injection disposal site would be similar to those associated with any heavy engineering project. In considering these impacts, it must be remembered, however, that the disposal site would include reprocessing and disposal facilities.

Ecological impacts from these processes are categorized into preconstruction and post-construction activities. Initial construction activities would involve clearing vegetation, drilling, and geophysical surveying. Impacts of these initial activities would affect vegetation, soil, water, and other resources to varying degrees depending on the characteristics of the specific site being developed. Impacts of this type of activity are evaluated for specific sites.

Construction impacts would include those of a reprocessing facility, as described in Chapter 4. Construction of facilities to prepare the wastes for injection, as described above, would also be needed.

Postconstruction, or operational, nonradiological ecological impacts would be more limited than those of preconstruction and construction activities. Many operational activities would occur below the surface. Ecological impacts from these activities could occur if some of the fluids injected into the well were to enter the ground-water system and were transported to the biosphere or otherwise affected aquatic resources. Surface runoff or material spilled on the surface could also cause localized ecological impacts.

Socioeconomic Impacts

Socioeconomic effects from constructing and operating a well injection repository would be felt most intensely in the immediate vicinity of the facility. In general, impacts would be representative of those of a major engineering facility. No quantitative data exist on the construction or operational employment requirements of a well injection disposal system. Impacts, however, should be similar to those described for the very deep hole concept (see Section 6.1.1.6). In addition, socioeconomic impacts associated with the reprocessing facility would be felt at the disposal site. These impacts are discussed in Section 4.7. In analyzing these discussions, it must be remembered that colocation would lead to a greater concentration of impacts at the disposal site, but at the same time would reduce the number of separate nuclear facilities constructed.

Aesthetic Impacts

Aesthetic impacts for the well injection disposal option would be similar to those of other subsurface disposal methods except for the presence of the reprocessing facility at the disposal site. Again, collocating facilities could increase the impacts at the chosen site, but the fact that only one site is needed suggests an overall reduction in aesthetic impacts.

Aesthetic impacts could be accurately assessed only within the context of a specific site. In a general context, however, aesthetic impacts related to drilling and other geologic activities are covered in the aesthetic impact discussions for mined geologic

repositories (Section 5.5) and the very deep hole concept (Section 6.1.1.6). Aesthetic impacts of reprocessing facilities are discussed in Section 4.7.

Resource Consumption

Suitable well injection sites would be sedimentary basins, which are frequently prime areas for fossil fuels. However, after the wastes had been safely emplaced, geologic exploratory activities in the vicinity of the site would have to be restricted. It has been suggested that potentially usable minerals from the zone of influence of the repository would be inventoried before implementation would begin. On the other hand, the disposal zone itself could be considered a resource for which alternative uses might be found, for example, storage of freshwater or natural gas.

Other resources consumed in the well injection process would include energy for transportation, processing, and disposal. Land would be required for the reprocessing and disposal facilities. For the shale-grout disposal method, clay, cement, and other materials would be needed. No critical material, other than fuel, would be consumed by well injection disposal.

International and Domestic Legal and Institutional Considerations

Implementation of the well injection option would require two important policy decisions that could be shaped by institutional forces. First, the process does not lend itself to handling spent fuel from reactors. Processing would be needed to transform this material into a form that could be readily injected into the well. The reprocessing approach most often proposed contravenes the current U.S. position against reprocessing. This would have to be resolved before well injection disposal could be implemented.

The second policy decision stems from the need to locate the disposal facility and the fuel reprocessing plant at the same site. Although such a system would be effective in limiting liquid waste transportation, it is likely that neither facility would be optimally located. It would have to be decided whether the benefits of well injection disposal outweigh potential disadvantages of such collocation. Obviously, such a decision would have to be made in light of domestic institutional considerations.

Another aspect of the well injection concept that could foster concern is the need to obtain records of previous drilling activities. States typically maintain such records and generally oversee drilling programs. If this disposal option were implemented, information would be needed and procedures would have to be established to evaluate data from adjacent well sites. The relationship between existing regulatory activities and the well injection disposal process would have to be defined prior to implementation.

Aside from the issues outlined above, the legal and institutional considerations of this option would be similar to those of the mined geologic repository discussed in Section 5.5.

6.1.6.5 Potential Impacts Over Long Term (Postemplacement)

An unavoidable long-term impact of well injection waste disposal is that alternative storage or disposal applications for the site are eliminated. Examples of possible uses are

natural gas storage, freshwater storage, and disposal of other wastes of lower or shorter-lived toxicity. In addition, as noted earlier, exploration for natural resources and subsequent mining in a large area around the disposal facility would be subject to control. The extent of exclusion and limited activity buffer zones would depend on the characteristics of the disposal formation, and in particular, its hydrologic and geochemical conditions. Finally, evidence exists that injection of wastes into certain formations could potentially lead to seismic activity and earthquakes.

Potential Events

Natural Events. The long-term leaching and transportation of radionuclides in the ground-water system to the biosphere would be a fundamental pathway in the well injection concept, as it is with all geologic concepts. Assessment of the environmental impact would require predictive modeling of the rock mechanics, hydrology, and geochemistry of the disposal and containment formations, together with an adequate data base to characterize the biosphere. The disposal area would be selected to minimize the risks from seismic and volcanic activities and their effect on the hydrologic regime. Seismic events could induce tectonic effects within the disposal area, causing permeability and flow changes. Volcanic activity could result in catastrophic breach of the containment formation, or could generate unacceptable, thermally induced flow patterns. The risk of meteorite impact would be similar to that for a mined geologic repository; however, with deep-well liquid disposal, the waste would be in a more mobile form. The impact of gross changes, such as climate variations or polar ice melting, would, in general, depend on their effect on the hydrologic regime. Increased erosion (because of glaciation, for example) could reduce the cover of the disposal formation.

An impact of potentially major significance is the increased chance of an earthquake that could result from injecting waste material into rock formations. A relationship between deep well liquid injection and increased seismicity has been suggested (Evans 1966) in connection with earthquakes at Denver and injection at the Rocky Mountain Arsenal well. Other studies (Hollister and Weimer 1968, Dieterich et al. 1972) have shown that deep well injections in the Rocky Mountain Arsenal Range have been instrumental in producing seismic events. Obviously, such concerns are significant and would have to be seriously evaluated for specific sites. Knowledge of the in situ stress state for both concepts would be needed before proceeding with the well injection option because of the chance of earthquakes developing. The depth of shale grout injection would be limited by the requirement that vertical stresses be less than horizontal stresses.

Manmade Events. Exclusion and controlled-use buffer zones would be set up around an injection facility. Nevertheless, the risks associated with drilling into a waste-liquid or grout disposal formation would have to be considered. Changes in the surface and subsurface hydrologic regime of the area, because of reservoir construction, deep excavation and construction, and resource exploitation outside the buffer zone, would require analysis.

The geologic formation in which a well injection repository would be located would have to be bounded by impermeable strata and free of water-transmitting faults. Such formations occur in the sedimentary basins in the U.S., and it is these basins that oil and gas companies are exploring for petroleum and natural gas. This exploration could cause a major safety problem by connecting waste disposal zones with aquifers.

Potential Impacts

As with the mined geologic repository, the principal pathway for release of radionuclides to the biosphere in the long term would be by ground-water transport. It is believed, however, that the likelihood of ground water reaching the injected waste is extremely small.

The only quantitative estimates on the movement of radionuclides via ground water transport are from ORNL's experience with grout injection of remotely handled TRU waste into shale (ERDA 1977).

The maximum quantity of activity that could be leached from a single grout sheet was calculated, using data presently available (ERDA 1977). This sheet would have a volume of about 28,300 m³ (1 million ft³) and could contain as much as 500,000 Ci of ⁹⁰Sr (if a maximum waste concentration of 5 Ci/gal is assumed) and an equal amount of ¹³⁷Cs. Leach data indicate that the 6-month leach rate of radionuclides from cured grouts would not exceed 6.2 x 10⁻⁵ Ci/month of ¹³⁷Cs per sq ft of leached area, 1.7 x 10⁻³ Ci/month-ft² of ⁹⁰Sr, 5.5 x 10⁻⁷ Ci/month-ft² of ²⁴⁴Cm, and 5.6 x 10⁻¹⁰ Ci/month-ft² of ²³⁹Pu. If the entire grout sheet surface were exposed to water flow, a maximum of 62 Ci/ month of ¹³⁷Cs, 1700 Ci/month of ⁹⁰Sr, 0.6 Ci/ month of ²⁴⁴Cm, and 6 x 10⁻⁴ Ci/ month of ²³⁹Pu would be leached. If the water flow is assumed to be 0.5 ft/day, the calculated concentration of ²³⁹Pu in the water would be approximately 1 x 10⁻⁶ Ci/ml (less than the concentration guide for this isotope in uncontrolled areas). The shale surrounding the grout sheets has considerable ion-exchange capacity for cesium and strontium; a calculation yields rate of movement of leached cesium and strontium through the shale that would be so low that these nuclides would be transmuted by radioactive decay long before they approached the surface. The small quantity of ²⁴⁴Cm that might be leached would also be retained by the shale.

6.1.6.6 Cost Analysis

Capital, operating, and decommissioning costs of well injection disposal have not been estimated. However, since well injection disposal would not require costly mining operations, it could offer a low-cost means of disposal compared to mined repositories.

Cost data are available from ORNL (ERDA 1977) for a site-specific application of grout injection disposal of RH-TRU. Estimated capital costs for a new waste shale fracturing disposal facility, adjusted to 1978 dollars, are \$6.0 million. Annual operating costs are estimated at \$110,000. No data are given for decommissioning costs. The costs are estimated

for a facility to perform removal of large volumes of mobile radioactive wastes from existing near-surface storage facilities at Oak Ridge.

6.1.6.7 Safeguard Requirements

Because of the restrictions concerning the transportation of high-level liquid waste, which require the injection facility to be collocated with the fuel reprocessing plant, the accessibility to sensitive materials would be extremely limited. However, this waste disposal system would probably be used in a uranium-plutonium recycle fuel cycle so there would be incremental increases in accessibility in other parts of the fuel cycle similar to most recycle scenarios. In addition, the difficulty of retrieving material once it had been successfully disposed of would increase the difficulty of diversion and the waste form (liquid) would complicate the transportation and handling problems for a potential diverter. The deep well injection repository would require additional safeguards since at least partial retrieval by drilling and pumping might be possible. Material accountability would be enhanced by ease of sampling and measurement of liquids, but gross accountability (i.e., gallons vs canisters) would be slightly more difficult than for the reference mined geologic concept.

See Section 4.10 for additional discussion of predisposal operations safeguard requirements.

6.1.7 Transmutation

6.1.7.1 Concept Summary

The primary goal of waste disposal has been stated as protection of the public. This would be achieved in mined geologic disposal by containing the high-level radioactive waste for the time period during which it retains significant quantities of potentially harmful radionuclides. One alternative to this approach is to selectively eliminate the long-lived radionuclides by converting or transmuting them to stable or short-lived isotopes. This approach would shorten the required containment period for the remaining waste. Shortening the containment period would increase confidence in predicting the behavior of the geologic media and reduce the requirements on the isolation mechanism. Thus, an attractive feature of transmutation is that it has the potential to reduce the long-term risk to the public posed by long-lived radionuclides.

In the reference transmutation concept, spent fuel is reprocessed to recover the uranium and plutonium. The remaining high-level waste stream is partitioned into an actinide stream and a fission product stream. The fission product stream is concentrated, solidified, vitrified, and sent to a terrestrial repository for disposal. In addition, actinides are partitioned from the TRU-contaminated process waste streams from both the fuel reprocessing plant and the mixed oxides fuel fabrication plant. The waste actinide stream is combined with recycled uranium and plutonium, fabricated into fuel rods, and reinserted into the reactor. For each full power reactor year, about 5 to 7 percent of the recycled waste actinides are transmuted (fissioned) to stable or short-lived isotopes. These short-lived isotopes are separated out during the next recycle step for disposal in the repository. Numerous recycles result in nearly complete transmutation of the waste actinides.

A disposal system that uses transmutation would have the environmental and health impacts associated with the recycle of uranium and plutonium and with the partitioning of the actinides from the waste stream. If uranium and plutonium recycle were adopted for other reasons transmutation would be more feasible but would still involve additional impacts. For example, highly radioactive fuel elements containing recycled waste actinides would need to be fabricated, handled, and transported. The additional facilities and waste treatment processing steps required could be expected to increase effluent releases to the environment, the occupational exposure, the risk of accidents, and costs. Since only about 5 to 7 percent of the recycled waste actinides would be transmuted to stable isotopes in each reactor irradiation, numerous recycles would be required with attendant additional waste streams.

6.1.7.2 System and Facility Description

System Options

The reference concept was selected from several available options. These options are listed in Figure 6.1.20 for each major step in a flowsheet using transmutation.

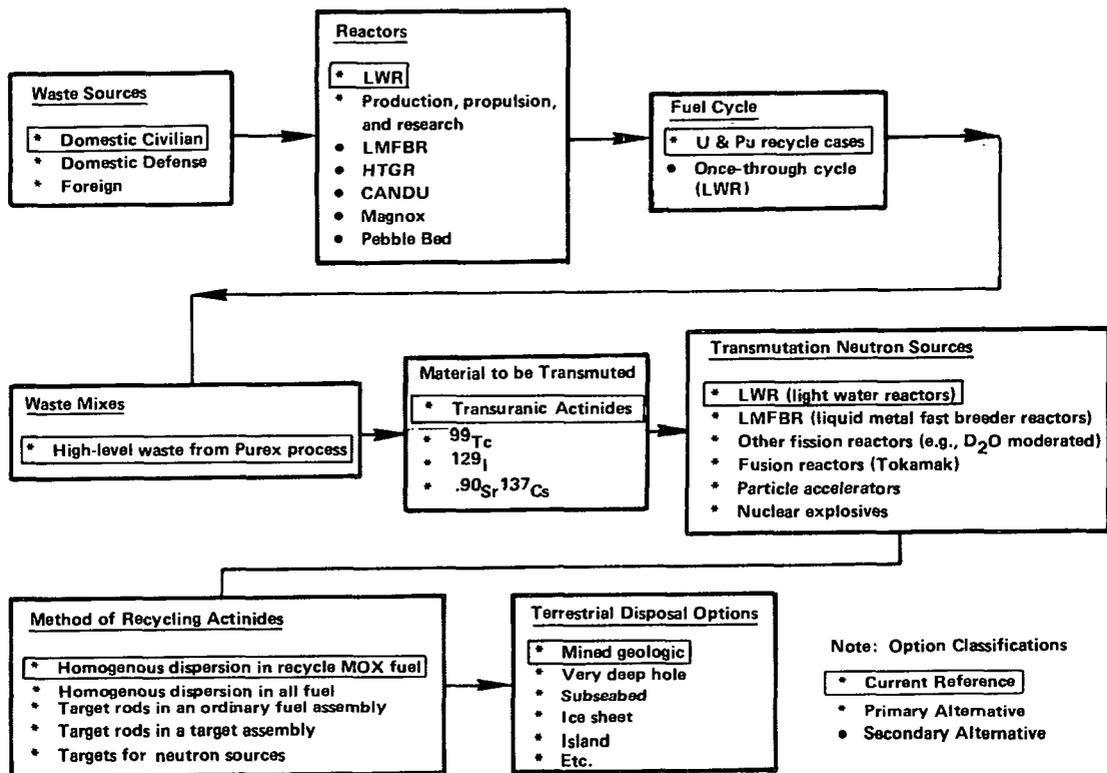


FIGURE 6.1.20. Major Options for a Waste Disposal Alternative Using Transmutation

The reference concept was selected somewhat arbitrarily to be used as a basis for comparison and to help identify the impacts associated with a typical transmutation fuel cycle. If transmutation were selected as a candidate alternative for further research and development, considerable study would be required to optimize the available alternatives. Additional information concerning the advantages and disadvantages of the many process options is available in sources listed in Appendix M.

Waste-Type Compatibility

Transmutation would be applicable to only those fuel cycles that involve the processing of irradiated nuclear fuel, e.g., the recycle of uranium and plutonium. In that context, transmutation would not apply to once-through fuel cycles. It could be used with both commercial and defense waste, although little work has been done concerning defense wastes.

Waste-System Description

The fuel cycle and process flow for the reference concept are shown in Figure 6.1.21. The cycle begins with the insertion of a reload of fuel into the reactor. The reload is two-thirds fresh enriched UO_2 and one-third recycle mixed oxide (MOX) fuel, which has all the waste actinides (i.e., neptunium and other transplutonics) homogeneously dispersed in it.

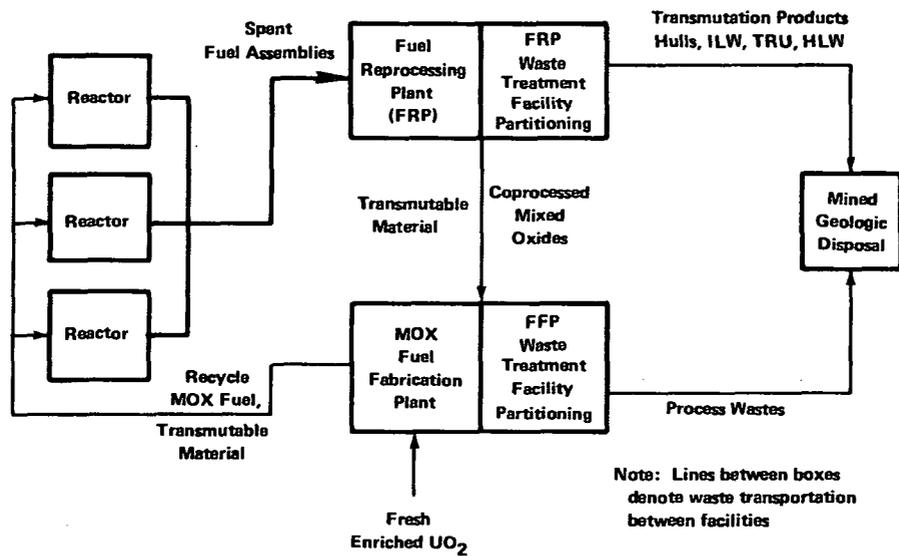


FIGURE 6.1.21. Partitioning-Transmutation Fuel Cycle Diagram

The cycle continues by:

- Irradiating the reload to a burnup of 33,000 Mwd/MTHM
- Discharging and decaying the reload for 1-1/2 years
- Reprocessing the UO₂ and MOX fuels together
- Sending the TRU-contaminated wastes to the fuel reprocessing plant waste treatment facility (FRP-WTF) for partitioning
- Returning the recovered TRU and the TRU-depleted wastes to the reprocessing plant
- Combining the recovered actinides with the processed MOX and transporting the mixture to the refabrication plant, after a 6-month delay
- Adding sufficient uranium to the MOX product to achieve the desired end-of-cycle reactivity. (This product is in powder form and contains the waste actinides.)
- Refabricating the MOX product
- Sending the TRU-contaminated wastes from refabrication to the fuel fabrication plant waste treatment facility (FFP-WTF) for partitioning
- Returning the stream of recovered actinides to the fabrication plant
- Incorporating the recovered actinides with MOX recycle streams within the facility
- Sending TRU-depleted wastes to a mined geologic repository.

Simultaneously, the fresh enriched UO₂ fuel is fabricated in a separate facility. At this point, the cycle is completed with the fabricated fuels being inserted into the reactor. The details of the waste treatment facility (WTF) process and plant design are given in Tedder et al. (1980) and Smith and Davis (1980).

Predisposal Treatment

In a fuel cycle involving transmutation, it would be necessary to partition the materials to be recycled and transmuted. The partitioning flowsheet would have two fundamental steps. The first would be to separate the actinides from other materials and the second would be to recover the actinides in a relatively pure form. Actinides would be separated by various methods and would originate from many sources, including high-level waste, dissolver solids, cladding, filters, incinerator ashes, salt wastes, and solvent cleanup wastes. The extractable actinides from these operations would be sent to actinide recovery, where they would be partitioned and purified.

Facilities Description

There are four facilities in the reference fuel cycle that process the actinides: the fuel reprocessing plant (FRP), the fuel fabrication plant (FFP), and a colocated waste treatment facility (WTF) for each. The purpose of the two WTF's would be to recover a high percentage of the actinides that would ordinarily be delegated to process wastes.

The FRP-WTF and FFP-WTF would have the following common process capabilities:

- (1) Actinide recovery
- (2) Cation exchange chromatography (CEC)
- (3) Acid and water recycle
- (4) Salt waste treatment
- (5) Solid alpha waste treatment.

In addition, the FRP-WTF would have high-level liquid waste and dissolver solid waste treatment process capabilities. The WTF facilities would be constructed on sites about 460 m (1,500 ft) from the FRP and FFP, but still within a fuel cycle center that would allow common services and utilities for the entire center. Additional detailed design and cost information is available in Smith and Davis (1980).

Since transmutation would take place in the reactor itself, no special facilities would be required, although the irradiation levels of the recycle fuel require that the fuel assemblies be handled remotely. Because transmutation would eliminate only a specific segment of the waste, all the facilities required for conventional terrestrial disposal, e.g., a mine geologic repository as described in Chapter 5, would also be necessary in this fuel cycle. The use of transmutation would not significantly change the total amount of waste or the necessary throughput of waste disposal facilities.

Retrievability/Recovery

The segment of waste disposed of in the mined geologic repository would exhibit the same characteristics discussed in Chapter 5 of this report.

6.1.7.3 Status of Technical Development and R&D Needs

Only the referenced use of transmutation - recycling, using commercial nuclear reactor fuels, to minimize the actinides contained in radioactive waste - is discussed here. Part of the R&D associated with transmutation would be the continued investigation of other useful applications of the process. There are several other waste constituents that could be transmuted.

Present Status of Development

Transmutation represents an advanced processing concept that would require R&D work before incorporation into any system. There are still uncertainties associated with many of the subsystem details. Although the concept is technically feasible, it should be recognized that the required design bases have not been sufficiently refined to permit construction of full-scale facilities. For some partition subsystems, laboratory experiments have been developed to demonstrate technical feasibility only. Only preliminary material balance calculations have been performed and, in most cases, no energy balances are available.

A number of transmutation devices for converting various nuclides to other more desirable forms have been studied. Neutron irradiation can be carried out with nuclear explosive devices, fission reactors, or fusion reactors. Accelerators can provide charged particle beams of protons or heavier ions for producing neutrons for irradiating selected nuclides. For the actinides, the most practical transmutation occurs by irradiation by a fission reactor neutron source. The estimated actinide transmutation rate utilizing commercial light water reactors is about 6 percent for each full-power year that the actinides are in the reactor (EPA/MITRE 1979).

There are four principal methods for recycling actinides in light water reactors: (1) dispersing the actinides homogeneously throughout the entire fuel reload, (2) dispersing the actinides homogeneously in only the mixed-oxide fuel, (3) concentrating the recycled waste actinides in target rods within an otherwise ordinary fuel assembly, and (4) concentrating the recycled waste actinides in target rods that are then used to make up a target assembly. In the first two methods, the actinides include all of the plutonium generated in the reactor. In the second two methods, plutonium (an actinide) is excluded from the targets but is recycled in a mixed-oxide fuel. On the basis of preliminary qualitative evaluation, it would appear that the second recycle mode, homogeneous dispersal of the actinides in the mixed-oxide fuel, is preferred over the others (Wachter and Croff 1980).

Technological Issues

The effect of a transmutation recycle, as opposed to the uranium and plutonium recycle mode, on the various elements of a conventional fuel cycle depends largely on two factors--the transmutation rate in the reactors and the manner in which the transmutation reactors are decommissioned as the cycle is eventually terminated. Important technological issues are:

- The use of commercial power reactors as transmutation devices might result in fissile penalties, reactor peaking problems, reduced reactor availability, and increased operating costs.
- Because of increased concentrations of radioisotopes with high specific activities, and/or modifications of existing systems due to changes in requirements, transmutation recycles could require additional containment systems to limit the release of radioactivity at the reactor site to acceptable levels.
- Many transmutation cycles would increase fuel handling requirements because of the more frequent insertion and removal of fuel and transmutation targets from the reactor core. Most transmutation cycles would result in increased shielding requirements both for fresh and spent fuels and transmutation targets.
- Decommissioning and disposal of reagents from partitioning and transmutation facilities would be complicated by the increased demands for shielding, multiple chemical processes, and waste streams.

The duration of the transmutation cycle is important in estimating its overall effectiveness in reducing the total radiotoxicity of transmutable elements in the environment. Premature termination of the transmutation cycle could actually increase the radiotoxicity of the wastes. This is because the resulting inventory sent to a final disposal system might have more activity than it would if the transmutation cycle had not been initiated.

R&D Requirements

The R&D requirements for partitioning would involve specific near-term subtasks to clarify points of uncertainty in the current process parameters and techniques. However, to fully develop and demonstrate actinide partitioning, a program would have to include additional process research and development, a cold (nonradioactive) testing facility, equipment development and testing, and pilot plant design, licensing, construction, testing, and operation.

Transmutation R&D would include specific nuclide cross section measurements, reactor physics calculations, and irradiation to full burnup of test fuel assemblies to verify calculations. The irradiation tests would also serve to confirm the design and fabrication of the fuel assemblies and their compatibility with and performance in the reactor during power operation.

The design, construction, and testing of a prototype shipping cask made from the relatively unconventional materials proposed might also be required. Specific aspects of cask technology that might require attention are: techniques for industrial fabrication of special shielding materials, such as B_4C/Cu and LiH , investigation of the ability of the cask using such materials to conduct the heat from the fuel contents, and the effect of the unusual construction materials on safety considerations in cask design.

Finally, continuing overall studies to define the preferred methods of operating the fuel cycle and the impacts and benefits of this operation would be of primary importance.

Implementation Time

The long lead time for implementing this alternative is based on the orderly development of a commercial scale partitioning plant, which would be expected to take about 20 years. The first 10 years would be devoted to partitioning research and the development and testing of a pilot plant, as reflected in Table 6.1.20. All of the R&D programs involving transmutation, fuel assembly and shipping cask development, and system studies could be accomplished in concurrence with the partitioning schedule.

Estimated R&D Costs

Table 6.1.20 identifies estimated R&D costs necessary to demonstrate the transmutation of actinides. It does not include costs associated with providing a commercial scale partitioning plant, the necessary modifications to the fuel fabrication facility and light water reactors, or a transportation system required to utilize the partitioning-transmutation of actinides as a waste disposal alternative.

Summary

Major uncertainties, shortcomings, and advantages of the concept are summarized below:

- The concept is actually a method of waste treatment or conversion to a more benign form; it is not an independent disposal method.
- Additional waste streams during the process are generated so that the actual volume of waste for isolation is greater than without it.
- The technology for efficient transmutation (waste partitioning and advanced reactors) are considered to be long-term achievements.

TABLE 6.1.20. Estimated Transmutation R&D Costs And Implementation Time

	<u>Cost, \$ million</u>	<u>Time Span, years</u>
Partition R&D (Includes Pilot Plant)	560	10
Transmutation R&D	16	15
Fuel R&D	80	15
Transportation	56	10
System Studies	8	Continuous

6.1.7.4 Impacts of Construction and Operation (Preemplacement)

As described in Section 6.1.7.1, the transmutation option would include elimination of certain long-lived radioactive wastes and the disposal of the remaining waste material in a mined geologic repository. The potential benefits of transmutation that would be realized for the lower levels of long-lived hazardous material are discussed in Section 6.1.7.5, while short-term impacts of construction and operation are discussed here. Because these short-term impacts include those of a mined geologic repository, impacts identified in Section 5.6 must be considered a part of this option. In addition, impacts associated with reprocessing and discussed in Section 4.7 would occur.

Because transmutation is a waste processing option involving extra waste treatment steps, a meaningful impact analysis is possible only when a transmutation system is compared with a reference processing and disposal system. In the following analysis, the reference system includes waste reprocessing and final disposal in a mined geologic repository.

Another important factor in this discussion is that impacts attributed to one plant generally relate to a reprocessing plant handling 2000 MTHM per year and a fuel fabrication plant handling 660 MTHM per year. Such a hypothetical plant provides the basis of much of the information used in this analysis (Blomeke et al. 1980, Fullwood and Jackson 1980, Logan et al. 1980). Depending on the actual amount of nuclear wastes generated, several of these plants could be constructed.

Health Impacts

Radiological Impacts. The increased frequency of waste handling and transportation activities associated with the transmutation option suggests that it would result in increased radiation exposures compared with the mined geologic repository option.

ORNL estimated the radiological occupational impact of the reference concept based on routine exposure, maintenance exposure, and anticipated abnormal occurrences (Fullwood and Jackson 1980). Table 6.1.21 presents the collective dose rates calculated for the four facilities included in the study. The values range from a low of 3 man-rem/plant-year for an abnormal occurrence in the FFP-WTF to a high of 230 man-rem/plant-year for routine and maintenance exposure in the FFP.

The radiological exposure to the general public arising from routine operations is a consequence of the fact that the facilities would have to provide fresh air for the workers and vent gases to the atmosphere. In spite of elaborate air-cleaning practices and equipment, small amounts of radioactive materials would be discharged into the atmosphere; the amount varying with the chemical species. Estimates have been made for the amounts of radioactive materials that are expected to be discharged from each plant (Fullwood and Jackson 1980). The resulting exposures, based on these estimates, are presented in Table 6.1.22. The values range from 680 to 736 man-rem/plant-year for the Reference Facility and the P-T respectively.

TABLE 6.1.21. Annual Routine Radiological Occupational Dose

Facility	Exposure, man-rem/plant-year		
	Routine	Operation Maintenance	Abnormal
FRP (1)	220	220	10
FRF-WTF (2)	220	220	10
FFP (3)	230	230	10
FFP-WTF (4)	90	90	3
Reference Facility (1) and (3)			
P-T	(1-4)		

The more significant of the postulated accidents have been analyzed as to the resulting effects on the plant workers. In general, individual worker exposure would exceed public exposure because of closeness to the accident. Isotopic differences between the two cycles would result in small differences in exposure, so there is negligible distinction between the Reference and the P-T cycle, except that the Reference Facility does not contain the two WTF's. The totals for the component facilities are presented in Table 6.1.23. The details of the accidents and other assumptions are given in Fullwood and Jackson (1980).

Table 6.1.24 presents corresponding data for the non-occupational consequences of the postulated accidents.

TABLE 6.1.22. Annual Routine Non-Occupational Dose

Process Stage	Exposure, man-rem/plant year	
	Ref. Facility	P-T
FRP	680	730
FRP-WTF	-	5.3
FFP	7×10^{-3}	1.7×10^{-2}
FFP-WTF	-	0.55
Totals	680	736

TABLE 6.1.23. Occupational Radiological Exposure--Abnormal

<u>Facility</u>	<u>Conditions</u>	<u>Exposure,</u> <u>man-rem/plant year</u>
	FRP	
FRP-WTF		1.3×10^{-2}
FFP		4×10^{-2}
FFP-WTF		7×10^{-3}

Besides the plants and processes another major activity in the fuel cycle would be transportation links for fresh fuel movement, spent fuel movement, powder movement between the FRP and FFP, and waste movement from the FRP-FFP complex to the repository and disposal area. Table 6.1.25 presents data resulting from accident analyses of the six transportation steps considered for the two fuel cycles.

Nonradiological Impacts. Nonradiological impacts would result from two factors that are unique to the transmutation alternative. First, the partitioning process would require additional facilities at the reprocessing plant and at the MOX fuel fabrication facility. Second, the nature of the wastes that would be generated by transmutation dictates increased transportation activities.

TABLE 6.1.24. Non-Occupational Radiological Exposures--Abnormal

<u>Process Stage</u>	<u>Exposure, man-rem/plant year</u>	
	<u>Ref. Facility</u>	<u>P-T</u>
FRP	5×10^{-3}	5×10^{-3}
FRP-WTF	-	6×10^{-5}
FFP	3×10^{-5}	3×10^{-5}
FFP-WTF	-	6×10^{-5}
Reference Facility	5×10^{-3}	
P-T		5.2×10^{-3}

TABLE 6.1.25. Transportation Non-Occupational Radiological Exposures--Abnormal

Transportation Step	Exposure, man-rem/plant year	
	Ref. Facility	P-T
Spent Fuel	2.3×10^{-3}	3×10^{-3}
Powder	2.3×10^{-10}	3×10^{-10}
Fresh Fuel	6×10^{-5}	3×10^{-5}
Cladding Hulls	1.2×10^{-2}	1.3×10^{-2}
HLW	8×10^{-4}	6×10^{-4}
NM-HLW	1×10^{-1}	9.8×10^{-2}
Totals	1.1×10^{-1}	1.1×10^{-1}

A closer examination of the first factor reveals that the additional partitioning facilities would be colocated at reprocessing and fuel fabrication sites. These incremental changes are analyzed as they would affect operational, environmental, and resource considerations.

Regarding the second factor, transportation impacts, the relatively small carrying capacity of the canisters that would be used to transport the fresh and spent fuel means more trips per unit of fuel than with options involving unpartitioned wastes. Furthermore, more waste would be generated. This would lead to more transportation impacts. It is estimated that the facilities included in this option would process 2,000 MTHM per plant per year. This means an estimated nine trips involving hazardous material would have to be made each day, as compared with an estimated seven trips per day for fuel reprocessing without transmutation (Fullwood and Jackson 1980). Although the increased emissions, chance of derailment, and community concern associated with more intensive transportation could not be accurately determined until a specific disposal system is proposed, it is recognized that transportation impacts would be greater than those for the reprocessing-only case.

Nonradiological health effects would occur as a result of construction and operation activities. In spite of scrubbers and other air-cleaning devices, small amounts of hazardous materials would be discharged into the atmosphere. There would be two main sources of these pollutants: the chemical processes themselves and the auxiliary services, primarily the steam supply system, which is assumed to burn fuel oil. Table 6.1.26 presents the annual health effects for transmutation. The data are based on estimates for the Allied General Nuclear services plant at Barnwell, South Carolina, but are scaled to allow for the larger size of the transmutation facilities. The health effects were estimated from epidemiological studies on SO₂ and its relationship to the other pollutants.

The increased transportation required for the transmutation alternative suggests a greater likelihood of occupational and nonoccupational hazards than with options not involving partitioning. Unlike radiological impacts, nonradiological concerns should not vary significantly from those of an industrial facility not involved in nuclear activity.

TABLE 6.1.26. Summary Effects (Per Plant-Year) of Non-Radiological Effluents (Fullwood and Jackson 1980)

Plant	Premature Deaths/yr		Permanent Disabilities/yr ^(a)	
	Reference Facility	Transmutation	Reference Facility	Transmutation
FRP	4	4	14	14
FRP-WTF	--	7	--	21
FFP	0.2	0.2	0.6	0.6
FFP-WTF	--	3	--	9
Totals	<u>4.2</u>	<u>14.2</u>	<u>14.6</u>	<u>44.6</u>

(a) Based on disabilities lasting longer than 6000 person-days.

Probably the single most important nonradiological hazard would result from the chemical processing, handling, and transportation activities, during which accidents could happen. The uncertainties associated with this unproven technology make precise analyses of these hazards difficult. Health evaluations, however, suggest that such hazards would pose approximately 20 times the risk of the radiological occupational hazards (Blomeke et al. 1980).

Other factors, such as seismic activity, fires, or severe meteorologic conditions, could lead to abnormal conditions. No such factors or their ensuing impacts, however, have been identified as warranting detailed environmental analysis for the transmutation facilities.

Natural System Impacts

Transmutation activity would involve handling several chemicals posing a potential health hazard. These chemicals would represent a threat to the natural environment surrounding fuel handling and processing facilities, as well as to the interconnecting transportation networks. Individual impact scenarios have not been postulated, but it can be assumed that there would be a risk of nonradiological impact associated with use of these chemicals not unlike that experienced by certain chemical process industries today.

Other nonradiological ecosystem impacts would result from construction, operation, and maintenance activities. Such impacts cannot be fully addressed except for a specific site. In general, potential impact would be similar to that of a comparably sized industrial operation. Reductions in the quantities of natural vegetation, an increase in runoff, and elimination of certain habitats are types of impacts that would be expected from such a facility. Although similar to impacts described for the baseline case of a fuel reprocessing operation that includes a mined geologic repository, the transmutation impacts would be greater because additional facilities and increased transportation would be involved.

Socioeconomic Impacts

Socioeconomic impacts associated with the transmutation alternative would occur primarily as a result of construction, operation, and transportation activities. Implementation of this alternative would involve a major construction force of over 3,000 individuals. Employment needs during operation would diminish to approximately 350 individuals per year for the FRP-WTF and 250 for the FFP-WTE (Smith and Davis 1980). These activities would also support increased transportation employment.

Compared to the baseline case of reprocessing without partitioning, operational employment levels for transmutation would increase substantially at the reprocessing and MOX fuel fabrication centers. Estimated work force increases are 35 and 80 percent at reprocessing and fuel fabrication facilities, respectively. Estimated socioeconomic impacts of such facilities are only conjectural at this point and specific impacts of hypothetical communities and groups are not included in this discussion.

Aesthetic Impacts

No data exist suggesting that aesthetic concerns from facilities required for transmutation activities would be greater than those associated with the reprocessing without partitioning. Neither the appearance or noise levels produced from the additional partitioning facilities should vary significantly from the baseline fuel reprocessing and preparation facilities.

Resource Consumption

Fuel and raw materials used in construction, as well as the chemicals and fuel required during operations and subsequent transportation activities, would be the most important resources used in the partitioning and transmutation process. For construction activities, a range of energy sources would be used in hardware fabrication and in actual construction operations. Other building materials such as steel, sand, and gravel typically used in major construction activities would also be consumed.

The reprocessing and partitioning process would also require quantities of chemicals, including nitric acid, hydrofluoric acid, hexanitrate acid, and several solvents. These chemicals would react with the waste material to form secondary wastes, as well as the desired end products.

Additional land would be required for this alternative. Facilities at the reprocessing plant should occupy 70 ha (172 acres) (Smith and Davis 1980) compared with 36 ha (90 acres) at present (DOE 1979c), and at the fuel fabrication plant 24 ha (59 acres) (Smith and Davis 1980) compared with 3 ha (8 acres) at present (DOE 1979c). Such a facility would normally process approximately 400 MTHM/year. In addition to the acreage occupied by each facility, large "restricted" areas would have to be established. Because of the conceptual nature of these facilities and the many possible ways they might be laid out, there are no specific estimates of the total size of restricted areas. At a minimum, the combined reprocessing and

waste treatment facility would require a 2400 ha (6000-acre) restricted area while the fuel fabrication plant would require a 4000-ha (10,000-acre) restricted area. These figures are based on estimates for the reprocessing and fuel fabrication plants without waste treatment facilities (DOE 1979c).

International and Domestic Legal and Institutional Considerations

The primary institutional concern associated with implementation of a transmutation process would be the compatibility between such a system and existing power reactors. Specifically, the use of commercial power reactors as transmutation devices might result in significant fissile penalties, reactor peaking problems, reduced reactor availability, shielding requirements for fresh fuel, increased operating costs, and the need for significantly more enriched ^{235}U as a driver fuel. Consequently, technological improvements in transmutation processes or an evaluation of the institutional framework surrounding establishment of new nuclear plant operating standards is needed before the transmutation alternative can be implemented.

Finally, it must be recognized that the partitioning and transmutation processes include intensive reprocessing of nuclear waste material and plutonium recycle. Adoption of the transmutation alternative therefore, would be inconsistent with this nation's current policy regarding reprocessing.

6.1.7.5 Potential Impacts Over the Long Term (Postemplacement)

Successful implementation of the transmutation process would reduce the long-term hazards associated with waste material. In fact, effective transmutation would virtually eliminate concerns with actinides and their daughters. Although the potential long-term benefits would be significant, there are long-term uncertainties and problems that must be weighed against them.

Potential Events

For this option, TRU-depleted wastes are assumed to be sent to a mined geologic repository. Therefore, events leading to potential problems over the long term for this option would be the same as those associated with the mined geologic repository (see Section 5.6). A major difference exists in impacts, however, because transmutation wastes would not be as toxic in the long term (beyond 1,000 years).

Potential Impacts

Impacts over the long term would be expected to be less severe than those anticipated with reprocessing only, since the waste placed in the repository would be partitioned and transmuted to reduce its toxicity. An important exception to this would occur following early termination of the transmutation cycle. Such termination can actually increase the radiotoxicity of the wastes, as mentioned earlier (Croff et al. 1977).

Results of a long-term risk comparison (Logan et al. 1980) between a reference (no transmutation) and a transmutation fuel cycle indicate that:

- Cs-137 and Sr-90 would dominate the health effects during the first few hundred years for both fuel cycles.
- After a few hundred years and for several tens of thousands of years thereafter, the most significant nuclides for the reference fuel cycle would include a generous mix of actinides and their daughters at a significantly reduced activity level. Transmutation would strongly reduce the effects during this period.
- During later years, two nuclides, Tc-99 and I-129, which are released by leaching, would completely dominate all other nuclide contributions. Because these nuclides are not removed through transmutation, the results show no benefit during these later years.

Long-term health effects have been integrated over 1 million years to determine the long-term probabilistic (expected) risk (Blomeke et al. 1980 and Logan et al. 1980). The long-term risk was found to be controlled to a very large extent by the contributions from Tc-99 and I-129, which constitute about 99 percent of the integrated risk. This is because (1) the slow leach incident dominates the long-term probabilistic risk since it was assumed to have a much higher probability of occurrence than a volcanic or meteor incident and (2) only those nuclides that sorb poorly or not at all (i.e., iodine, technetium, carbon) migrate through the geosphere quickly enough to reach the biosphere within 1 million years. Therefore, transmutation of actinides would have its most substantial value if an unlikely event occurs. For example, the probability of a volcanic incident is only one in 100 billion, but if it should occur, the radioactive material could enter the biosphere very rapidly.

Looking at the issue described above in another way, it is noteworthy that catastrophic events occurring beyond 100 years following emplacement would not cause significant radiologic health effects if transmutation were employed.

6.1.7.6 Cost Analysis

The cost of utilizing transmutation to modify the radionuclide composition of waste would be added to the cost of disposal associated with remaining modified waste. However, modification of the waste's radionuclide content has the potential to alleviate some of the disposal requirements and reduce these costs. Such costs have not been developed at this time.

Costs have been developed for a fuel cycle including actinide transmutation utilizing commercial light water reactors as the transmutation device. These were compared with the costs of a mixed-oxide fuel cycle (Alexander and Croff 1980). This study indicated cost increase of about 3 percent for nuclear generated electricity if actinide transmutation were utilized for disposal purposes.

The significant cost differentials were associated with the requirement of specialized partitioning facilities and hardware. The continued recycle of actinides into the fuel cycle would increase the neutron activity within the fuel material about tenfold for spent fuel and

more than 100 times for fresh fuel. These increases must be taken into account by increased shielding and by use of remote operations and maintenance when designing fuel cycle facilities. Reprocessing costs would increase by an estimated 5 percent, fuel fabrication costs would double, and transportation costs would nearly triple (Smith and Davis 1980).

The following cost estimates are for only the specialized partitioning facilities collocated with their respective mixed-oxide fuel fabrication facility and spent fuel reprocessing facility. The fuel fabrication plant has a throughput of 660 MTHM per year and the reprocessing plant a throughput of 2,000 MTHM per year.

Capital Costs

The partitioning process buildings are first-of-a-kind facilities that, in several instances, include process operations that have not advanced beyond laboratory test and evaluation. Therefore, considerable judgment was used in the development of the capital costs shown in Table 6.1.27.

Operating Costs

Estimated operating costs are shown in Table 6.1.28. Labor cost estimates are based on an average salary of \$20,000 per year for management, engineering, and supervision and \$14,500 per year for operators, maintenance personnel, guards, laboratory technicians, and clerical personnel.

TABLE 6.1.27. Capital Costs For Partitioning Facilities
(Millions of 1978 Dollars)
(Smith and Davis 1978)

	Colocated With Reprocessing Plant			Colocated With Fuel Fabrication Plant		
	Material	Labor	Total	Material	Labor	Total
Land Improvements	1.3	1.2	2.5	1.0	.9	1.9
Process Facilities	200.0	127.0	327.0	73.1	46.9	120.0
Tunnel and Piping	5.8	10.6	16.4	4.9	9.2	14.1
Support Facilities	13.0	5.7	18.7	12.2	4.6	16.8
Subtotal	220.	145	365.	91	62	153
Field Indirects and S/C's OH&P			145			62
Subtotal			510			215
Engineering & Design			143			60
Subtotal			653			275
Contingency			228			96
Total			881			371

TABLE 6.1.28. Operating Costs For Partitioning Facilities
(Millions of 1980 Dollars)

	<u>Colocated With Reprocessing Plant</u>	<u>Colocated With Fuel Fabrication Plant</u>
Process Chemicals	16.0	1.4
Utilities	6.2	2.2
Labor	8.2	5.8
Equipment Replacement	3.8	1.0
Property Tax and Insurance	26.0	11.1
NRC License and Inspection	<u>0.2</u>	<u>0.2</u>
Total	60.4	21.7

Decommissioning

Decommissioning costs associated with the partitioning facilities were estimated to be 12 percent of the capital costs for the partitioning facilities, i.e., \$105 million for the facility colocated with the reprocessing plant or \$45 million for the facility colocated with the fuel fabrication plant.

6.1.7.7. Safeguard Requirements

The transmutation concept depends on processing of the spent fuel elements and the recycle of transmutable materials. The extra processing and transportation, and the availability of sensitive materials at all points in the back end of the fuel cycle would increase the opportunity for diversion of these materials. In addition, because of the necessity to process and recycle material eight or nine times to ensure full transmutation, the annual throughput of sensitive materials would greatly increase. Material accountability would also be more difficult because of the large quantities and high irradiation levels. Safeguards of recycled plutonium would be simplified because of the higher concentration of ^{238}Pu . Also, recycled actinides containing ^{252}Cf and ^{245}Cm would require shielding from neutrons that should simplify safeguard requirements. Furthermore, because geologic disposal would be required on the same scale as discussed in Chapter 5, all the safeguard requirements described there would also be required for a fuel cycle using transmutation. See Section 4.10 for additional discussion of predisposal operation safeguard requirements.

6.1.8 Space Disposal

6.1.8.1 Concept Summary

Space disposal offers the option of permanently removing part of the nuclear wastes from the Earth's environment. In this concept, HLW would be formed into a cermet matrix and packaged in special flight containers for insertion into a solar orbit, where it would remain for at least 1 million years. NASA has studied several space disposal options since the early 1970s. A reference concept using an uprated Space Shuttle has emerged and is considered in detail here.

The Space Shuttle would carry the waste package to a low-earth orbit. A transfer vehicle would then separate from the Shuttle to place the waste package and another propulsion stage into an earth escape trajectory. The transfer vehicle would return to the Shuttle while the remaining rocket stage inserted the waste into a solar orbit.

The space disposal option appears feasible for selected long-lived waste fractions, or even for the total amount of high-level waste that will be produced. The remaining TRU wastes would require some terrestrial disposal option, such as mined geological repositories in the continental U.S. Space disposal of unprocessed fuel rods does not appear economically feasible or practical because of the large number of flights involved.

Space disposal was considered for its potential to reduce long-term environmental impacts and human health effects for a given quantity and type of waste compared with alternative terrestrial disposal options. Because of the characteristics of the space disposal concept, which removes the waste package from the biosphere, it is highly unlikely that physical forces would cause the radioisotopes to migrate toward the Earth. Consequently, for a package properly placed in orbit, there would be no long-term risk or surveillance problem as in terrestrial alternatives. However, the risk and consequence of launch pad accident and low earth orbit failure must be compared to the risk of breach of deep geologic repositories.

6.1.8.2 System and Facility Descriptions

System Options

The reference concept and system for the initial space disposal of nuclear waste has been developed from a number of options available at each step from the reactor to ultimate space disposal. These options are summarized in Figure 6.1.22 (Battelle 1980), which indicates currently preferred options chosen for the DOE/NASA concept, primary alternatives, secondary alternatives, and options that are no longer considered viable. The bases for selection of options for the reference concept (those blocked off) are detailed in various sources listed in Appendix M.

Waste-Type Compatibility

As noted, space disposal of unprocessed spent fuel rods would be impractical because an excessive number of launches would be required. This would result in high energy re-

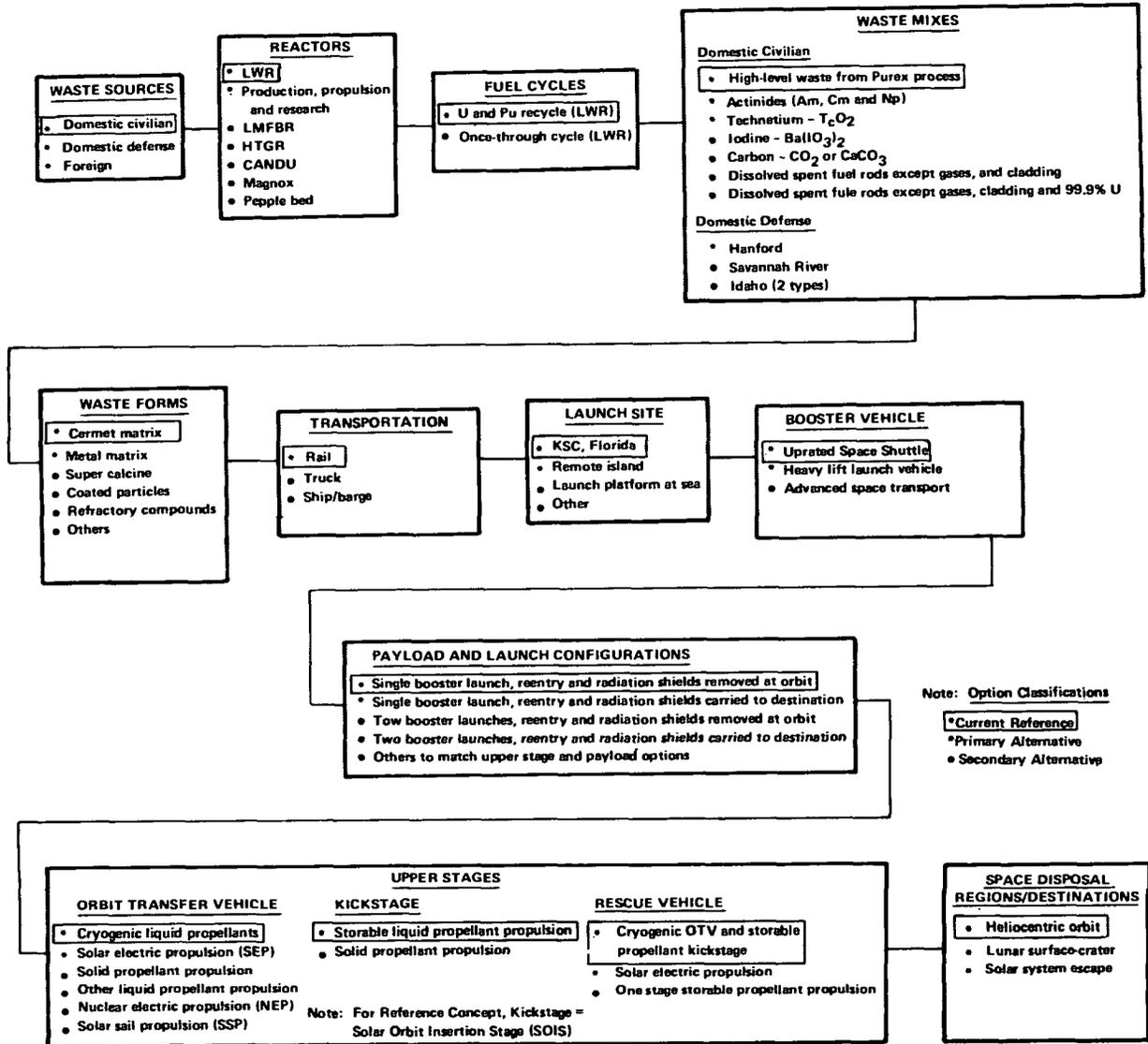


FIGURE 6.1.22. Major Options for Space Disposal of Nuclear Waste

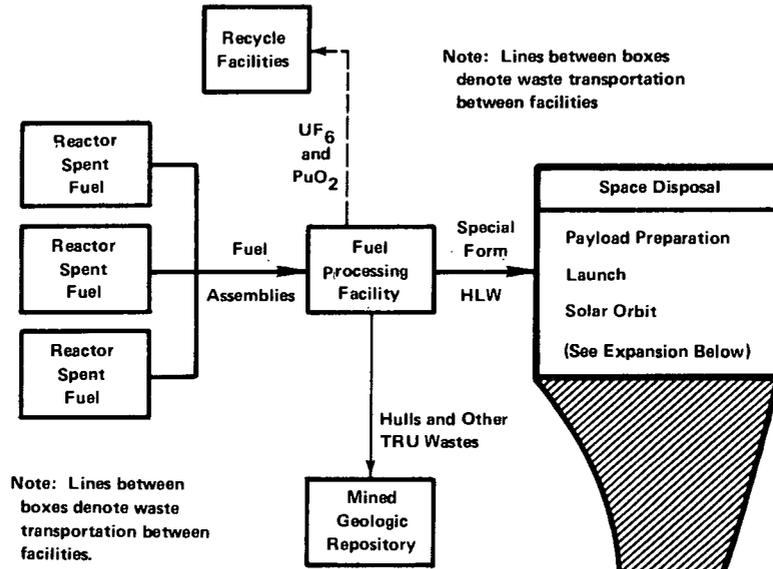
quirements, high costs, and probably increased environmental impacts (see Section 6.1.8.4). Thus, some form of waste separation would be required. For HLW, the option appears to be feasible, on the basis of the much lower number of Space Shuttle flights that would be required (approximately one launch per week to dispose of HLW from 5000 MT of heavy metal resulting from operations of approximately 170 GWe nuclear capacity). It is also possible that the space option would be used to rid the Earth of smaller quantities of radioactive wastes that pose special hazards for long-term terrestrial disposal. The disposal of selected isotopes would require chemical partitioning, with its high costs and secondary waste streams. Remotely handled and contact-handled TRU wastes from the recycle options would require geologic disposal.

Waste-System Description

The concept for space disposal of nuclear waste described here is the current DOE/NASA reference concept as reflected by the preferred options in Figure 6.1.22. To place the space disposal concept into perspective from a total system viewpoint, Figure 6.1.23 shows the waste management system, emphasizing the location and process flow details of the space disposal alternative within the total system. Two points are apparent from this figure: (1) chemical processing would definitely be required for space disposal of waste, and (2) the mined geologic repository would be part of the total system. The following discussion briefly summarizes the mission profile from the standpoint of waste-type compatibility, prelaunch activities, and orbital operations. Battelle (1980) presented a more detailed discussion of this profile and various element definitions and requirements.

Prelaunch Activities. The prelaunch activities would include nuclear waste processing and payload fabrication, ground transportation of waste, on-site payload preparation, and final staging operations.

Typically, spent fuel rods from domestic power plants would be transported to the waste processing and payload fabrication site in conventional shipping casks (see Chapter 4). A high-level waste stream containing fission products and actinides, including several tenths of a percent of the original plutonium and uranium, would result from the uranium and plutonium recovery process. This waste would be formed into a "cermet" matrix (Aaron et al. 1979) (an abbreviation for ceramic particles uniformly dispersed within a metallic phase), which has been shown to have superior properties compared with other potential waste forms for space disposal (Battelle 1980). The waste would then be fabricated into an unshielded 5000-kg sphere. Within a remote shielded cell, this waste payload would be loaded into a container, which would be closed, sealed, inspected, decontaminated, and packaged into a flight-weight gamma radiation shield assembly. During these operations and subsequent interim storage at the processing site, the waste package would be cooled by an auxiliary cooling system.



FUEL CYCLE DIAGRAM - SPACE DISPOSAL

HLW From Fuel Processing Facility

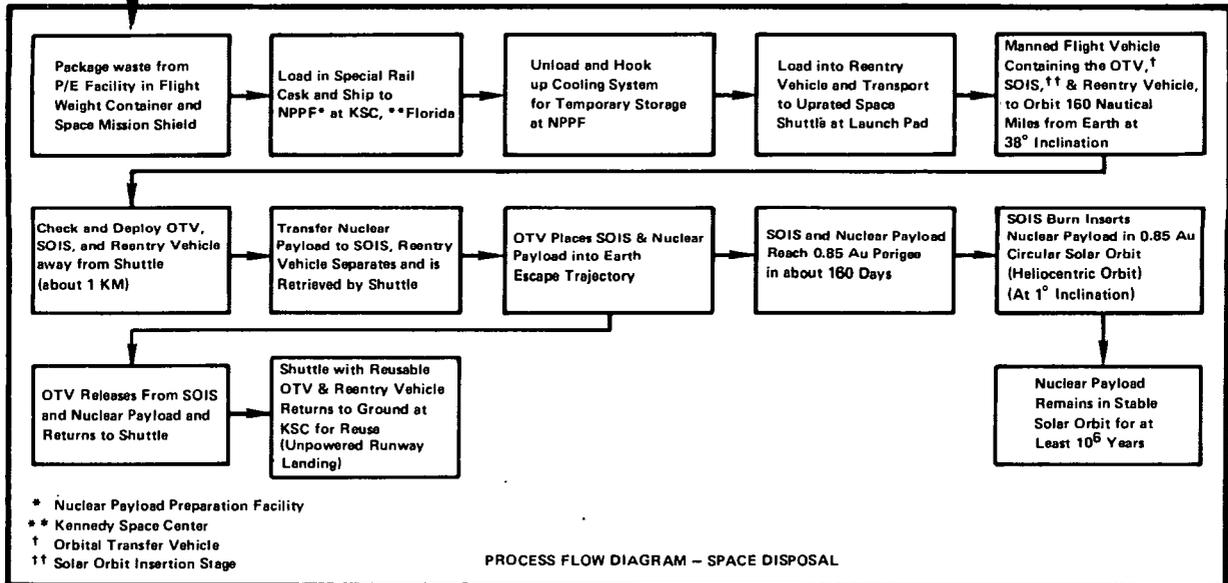


FIGURE 6.1.23. Waste Management System--Space Disposal

The shielded waste container would be loaded into a ground transportation shipping cask. This cask would provide additional radiation shielding, as well as thermal and impact protection for the waste container to comply with NRC/DOT shipping regulations. It would be transported to the launch site on a special rail car and be stored in a nuclear payload preparation facility with provision for additional shielding and thermal control. The waste containers would be monitored and inspected during storage.

For launch, the shielded waste form would be integrated with:

- A reentry vehicle, which would protect and structurally support the waste in the Space Shuttle orbiter cargo bay
- A solar orbit insertion stage (SOIS), which would place the waste payload into its final solar orbit
- An orbit transfer vehicle (OTV), which would take the waste from low Earth orbit into a solar orbit transfer trajectory.

Pre-launch checkout would include verification of the payload and the payload-to-orbiter interface systems. Typically, propellant would be loaded in the preparation facility to minimize the hazard of propellant loading while the payload was in the Shuttle cargo bay on the launch pad.

From the preparation facility, a special-purpose transporter would take the payload to the launch pad, where special equipment would position and install it in the Shuttle cargo bay.

Orbital Operations. The orbital operations for this concept would include launching into earth orbit, transfer from there to a solar orbit, and finally rounding out the solar orbit. (see Figure 6.1.24). The Up-rated Space Shuttle, designed to carry a 45,000 kg (99,000 lb) payload, would be launched into a low Earth orbit (300 km). The launch would avoid early land overflight of populated land masses. The liquid rocket booster engines and the external tank would be jettisoned before the orbit is reached.

During suborbital portions of the flight, the Orbiter would be able to command shutdown of all engines and either return to the launch site or ditch in the ocean. From 5 to 6 minutes after launch, the Orbiter could abort by going once around the Earth and then returning to land. After 6 minutes, the Orbiter has the on-board thrust capability to abort directly to a sustained earth orbit. If a Shuttle malfunction exceeded the abort capability, the nuclear payload with the reentry vehicle would automatically eject and make its own reentry. It would be designed to survive a land or water impact.

Once in orbit, the loaded reentry vehicle would be automatically latched to the SOIS and, with the OTV, would automatically deploy from the orbiter bay. At this time, the waste payload would be remotely transferred from the reentry vehicle to the SOIS payload adapter.

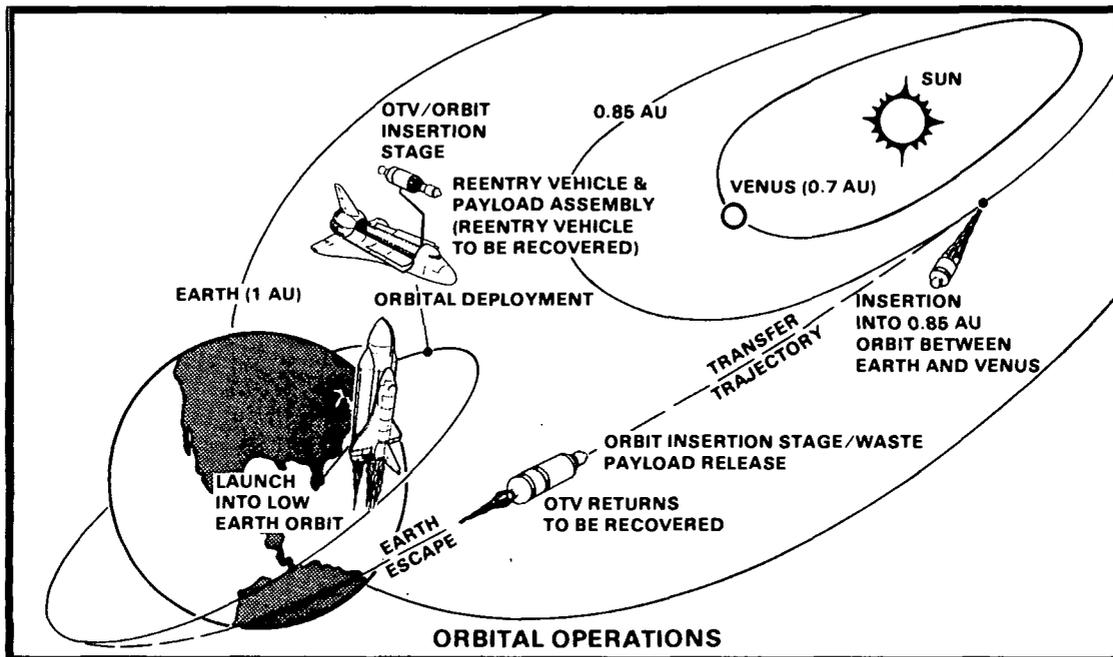


FIGURE 6.1.24. Orbital Operations

After a final systems checkout, the OTV would place the SOIS and its attached waste payload into an Earth escape trajectory. Propulsion would be controlled from the Orbiter, with backup provided by a ground control station. After propulsion, the OTV would release the SOIS/waste payload and would return to low Earth orbit for rendezvous with the Orbiter. The payload would require about 163 days to reach its perihelion at 0.85 astronomical units (A.U.) about the Sun. (One A.U. is equal to the average distance from the Earth to the Sun.) Calculations have shown that this orbit would be stable with respect to Earth and Venus for at least 1 million years.

In case of OTV ignition failure, a rescue OTV would be launched to meet and dock with the SOIS for propulsion into the escape trajectory. Safety features would be included in the design of this vehicle to prevent reentry of the unshielded payload into the Earth's atmosphere (Bechtel 1979a).

After rendezvous with the OTV, the Shuttle Orbiter would return to the launch site for refurbishment and use on a later flight. The empty reentry vehicle would also be recovered and returned with the Shuttle for reuse. The normal elapsed time from launch to return to the launch site would be 48 hours (Bechtel 1979a).

Systems for tracking the vehicles during launch, earth orbit, and the earth escape trajectory exist. There is also a system for locating and tracking the payload in deep space at any future time. However, once the proper disposal orbit had been verified, no additional tracking should be necessary.

Retrievability/Recovery. Until the waste package had been successfully disposed of in accordance with the design, retrieval or recovery capability would be necessary. A discussion of the rescue technology required for such a retrieval capability is presented in Section 6.1.8.3 below.

6.1.8.3. Status of Technical Development and R&D Needs

Present State of Development and Technological Issues

While the space option appears technically feasible, there are engineering problems that would require resolution. The Space Shuttle is currently in development and the first orbital flight is scheduled in 1981. The Space Transportation System should eventually (1990s) include a Space Shuttle with liquid rocket boosters (replacing current solid rocket boosters) and a reusable OTV. NASA has studied such vehicles extensively for future space missions and they represent a logical extension of the space transportation capability upon which to base a reference concept.

Many aspects of the space disposal system represent straightforward, applications of existing technology, e.g., use of liquid propellants and reentry vehicle design; however extensive engineering development would be required. The major technology development requirements are in design for safety, environmental impact analysis of space launches, and waste preparation. The nuclear waste payload container and reentry vehicle are only conceptually defined and additional study would be required to assure that safety and environmental requirements could be met in case of launch pad and reentry accidents. Development of a capability for deep space rendezvous and docking to correct improper orbit of a waste package would be required. The current status of development and research needs in specific areas are discussed below.

Emplacement Methods. The technology for launching both nuclear and nonnuclear payloads into space is highly developed, but the technology for putting nuclear waste in space is still in a conceptual stage. Earlier experience with space nuclear auxiliary power (SNAP) systems employing radioactive thermoelectric generators provides some experience, particularly in safety analyses, but the amounts of radioactive materials in such systems are much less than those that would be associated with waste payloads. The present DOE/NASA conceptual definition is based on technology and equipment used previously in other space missions but which would require design modifications for use in waste disposal missions. For example, the Space Shuttle power plant would need to be upgraded to increase payload capacity and thereby reduce the number of flights required. On the basis of the results obtained in the space program, considerable confidence has been gained in ability to design the necessary high-reliability systems. Procedures currently being developed to address abort contingencies for the manned Space Shuttle would be useful to mitigate adverse effects of aborts in waste launch operations.

Waste Form. The waste form would have to be a nondispersible, chemically stable solid. The composition of this waste has not been defined by the space program sponsors, but there are several possible candidate processes that might produce the proper form, as suggested in Figure 6.1.22.

The waste form should contribute to overall system safety, especially for potential accident sequences, and should also contribute to system optimization in terms of payload, economics, and materials compatibility. Desirable attributes are:

- High HLW to inert content ratio
- High thermal conductivity
- Resistance to thermal shock
- Thermochemical stability
- Toughness
- Low leachability
- Applicable to both commercial and defense wastes
- Resistance to oxidation
- Low cost
- Ease of fabrication.

Because weight would be important in the launching operation, the waste forms should also maximize the amount of waste carried at each launch (waste loading). An iron/nickel-based cermet prepared by ORNL for other disposal options appears suitable, but would require further development.

Waste Package. The reference waste package would consist of the spherical waste form surrounded by a metal cladding, a gamma shield, a steel honeycomb structure (for impact), insulation (for reentry), a graphite shield (for reentry), and the reentry vehicle itself, which would contain the waste during launch and Earth orbit in case of accident. Only conceptual definitions have been developed.

Waste Partitioning. Certain space option alternative concepts would be enhanced if specific isotopes were removed from the waste, e.g., strontium or cesium. Alternatively, space disposal might be more appropriate for certain species, e.g., iodine, technetium, the actinides, or all three. Technology development would be needed to provide these partitioning options.

Facilities. The size, capacity, and functional requirements of the nuclear payload preparation facility are not defined. Major design tasks remain before this facility could be developed.

Rescue Technology. Remote automated rendezvous and docking capabilities would probably be required for space disposal of radioactive waste. The HLW payload would require technology development to provide recovery capabilities for payloads in deep space, especially for uncontrollable and/or tumbling payloads. Also, it might be necessary to develop new technology for deep ocean recovery of aborted or reentrant payloads. Deep ocean recovery has been demonstrated on several recent projects, but any new, special capabilities to handle HLW payloads would need to be defined. Special equipment to recover reentrant payloads that touch down on land might also be required, although the technological challenge would probably not be as great.

R&D Requirements

In the final analysis, R&D needs would depend on the space disposal mission selected. The R&D requirements for this program would span the spectrum from systems definition conceptual studies through generic technology development (e.g., waste form) to engineering developments of facilities and hardware (e.g., the payload preparation facility and tailored space vehicles). These latter aspects would be deferred until the space disposal mission is better defined.

Thus, initial R&D would need to cover the following elements for concept definition and evaluations, listed approximately in sequential order.

- Perform trade-off and risk analysis studies to select the mix of radionuclides for space disposal
- Assess technology availability of waste processing and waste partitioning options
- Develop waste form criteria and options for space disposal
- Define facilities and ground transportation systems requiring R&D
- Define waste payload systems and containment requirements
- Define and select flight support systems for the space disposal option (e.g., shielding)
- Complete conceptual definition of unique launch site systems
- Assess advanced launch systems under development for space disposal applicability
- Define possible systems for transferring nuclear waste from Earth orbit and recovering failed payloads
- Characterize possible space destinations and missions
- Assess unique safety and environmental aspects of the space mission (e.g., launch pad fires and explosions affecting the waste package).

These conceptual studies would set the requirements for future R&D programs, if warranted. Other applicable ongoing R&D projects, e.g., concept definition of metal matrix waste forms and advanced launch system definition, would be pursued concurrently.

Implementation Time

With the space disposal mission currently in the concept definition and evaluation phase, meaningful predictions of the initial operational date are not possible. However, the present DOE/NASA concept depends on the availability of an OTV and the Up-rated Space Shuttle that have not been developed. This space disposal system could be operational possibly by the year 2000. Major sequential outputs that could be derived from conceptual studies are:

- Identification of viable alternative space systems concepts
- Identification of viable nuclear waste system concepts
- Selection of preferred concepts
- Selection of baseline concept
- Completion of baseline concept definition
- Generation of development plan

Estimated Development Costs

Development costs would depend largely on the specific space option approved. Also, once that option was defined, ongoing work oriented to other Shuttle and waste disposal options could be refocused on space disposal requirements. Examples are deep space rendezvous and docking techniques and waste form technology development. This would identify the incremental Shuttle and waste isolation program costs attributable to space disposal.

Thus, funding requirements for development of the space disposal option have not been well defined. It would generally be assumed that NASA would undertake the development of the required space components and DOE would develop the waste technology if the concept was pursued. It assumed that the approach would be on an incremental basis. This work would include R&D and identification of design development requirements for nuclear waste systems and space systems for disposal, domestic/international affairs studies, and impact assessments. The studies would provide a cost basis for further programmatic decision making.

Summary

Major uncertainties, shortcomings, and advantages of the concept are summarized below:

- The concept does not permit ready corrective action.
- The concept is susceptible to single mode (launch pad) failure, unless well-engineered multiple barriers are developed to protect the waste.
- Significant technology advances and equipment development will be required.
- Waste form and package concept development are in a very preliminary stage.
- The concepts usefulness would be limited to waste from reprocessing or further limited to selected isotopes.

6.1.8.4 Impacts of Construction and Operation (Preemplacement)

A space disposal approach must consider the total integrated system risk, i.e., the risks of launching wastes into space and the risks associated with the secondary waste streams generated by waste treatment, the fraction of waste that would have to go to terrestrial disposal, and the increase in system complexity. Hence, the short-term health and environmental impacts would likely be increased, while risks associated with those residual waste forms that remained on Earth for disposal in a mined geologic repository would likely be decreased. The environmental and health impacts associated with the latter consideration are expected to be less significant than those associated with total terrestrial disposal of HLW.

In the early years of a space disposal program, certain modifications would be required at Kennedy Space Center, assuming it was selected as the launch site. At the least, this would involve construction of a payload preparation facility. If the total Space Shuttle traffic (including all space missions) saturated the capability of shuttle facilities, then modifications, or even new facilities (e.g., launch pads), would be necessary. New construction activities would be designed to have the minimum adverse effect on the area. NASA has concluded that all potential nonradiological environmental impacts foreseen during normal operation of the Space Shuttle would be localized, brief, controllable, and of minimum severity (NASA 1978). Results of an evaluation of the incremental impacts of construction of facilities to accommodate waste disposal via the Shuttle and other environmental impacts of the space disposal program are presented below (Bechtel 1979a).

Health Impacts

Normal operation of facilities are not expected to cause any significant adverse health effects from either radiological or nonradiological sources. During abnormal operations (a reentry and burnup accident) the total population radiological dose could be quite large; although the estimated average individual dose would be very small.

Radiological Impacts. Health impacts from routine operations would be related primarily to planned release of radioactive and nonradioactive materials. Impacts to man from routine operations would be derived from three of the five operational phases: predisposal treatment and packaging (reprocessing), transportation, and emplacement.

No significant adverse health effects would be expected from normal operation of reprocessing facilities (NRC 1976). Incremental effects of additional processing to partition specific nuclides are not expected to change this conclusion.

Health effects caused by terrestrial transportation would be expected to be no different for space disposal than for other waste disposal options and are assumed to be similar to those for existing containers that have been reviewed for safety and licensed by regulatory agencies.

The estimated total occupational whole-body radiation dose from space disposal (the three operational phases plus the terrestrial repository for secondary waste) is 6340 man-rem/yr

(Bechtel 1979a). (See Table 6.1.30.) Of this dose, 1000 man-rem/yr derives from Space Shuttle-related activities. The nonoccupational dose is estimated at 180 man-rem/yr, with a negligible amount attributed to the Space Shuttle program.

Accidents may be classified by their location within the sequence of operations as associated with:

- Waste treatment
- Payload fabrication
- Payload ground transportation
- Handling and launch preparation
- Launch phases (suborbital)
- Orbital operations
- Postemplacement.

Within this sequence, many possible accidents that might be called "typical industrial" accidents can be identified. These are not discussed further because they (a) are not related directly to either the nuclear or space transportation aspects, (b) have negligible environmental impact, and (c) are no more probable (and in fact may be less probable) in this activity than in any industrial activity of similar magnitude. Of primary concern here are those accidents involving radioactive material, that would lead to the release and dispersion of the radioactive material into the environment. Waste treatment, payload fabrication, payload ground transportation and handling, and launch preparation for the space disposal option would be expected to be broadly similar to the same activities as employed for terrestrial disposal options. Thus, the possible accidents and accident consequences would also be similar (subject to some variation relating to the different nuclides that might be involved). Such accidents and their consequences are treated in Chapter 4 and are not further described here.

Certain types of accidents that might occur during the launch or orbital and post-emplacment operations would impose difficult environmental conditions on the payload. They could lead to the payload coming to rest in uncontrolled areas or to the release and dispersion of some of the radioactive waste. These accident types would include:

- Explosions
- Intense fires
- High-velocity impact
- Atmospheric reentry.

The payload and other mission hardware, as well as the procedures used to carry out the various operations, would be designed to

TABLE 6.1.30 Short Term (Preemployment) Radiological Impacts For
The Space Disposal Program Normal Operation

	Whole-Body Dose, man-rem/yr	
	<u>Occupational</u>	<u>Nonoccupational</u>
Waste Processing Facility	4100	90
Transportation	210	90
Repository (Secondary Waste)	1030	Neg.
Space NPPF	70	Neg.
Transporter/Launch Pad	150	
Shuttle	<u>780</u>	
	<u>6340</u>	<u>180</u>

- Minimize the probability of events leading to severe environments
- Provide, when possible, a contingency action to remove the payload from the threatening environment
- Maximize the probability that the waste payload containment will not be violated if subjected to the environment.

Two important types of accidents, both unique to the space disposal option, are:

- A catastrophic, on- or near-pad explosion and fire of the booster launch vehicle
- A high-altitude reentry and burnup of an unprotected nuclear waste container, with subsequent conversion of a certain fraction of the payload to submicron particles of metal oxides.

Aside from immediate possible casualties and the close-in physical effects from, for example, the on-pad explosion and fire, the environmental impact of overriding significance for these events would be possible radiation exposure to the general public. Edgecombe et al. (1978) provides preliminary data on environmental conditions around catastrophic launch-pad accidents.

Short-term risks might or might not be lower than those for terrestrial disposal options. However, for the space disposal option to be implemented, they would have to be at an acceptable level. Reliability data for systems would be required before a risk assessment could be made. Reliabilities of the booster vehicle, upper stages, and safety systems envisioned for the space disposal mission have not yet been determined by NASA, but are expected to be high.

Regarding on- or near-pad accidents, no precise estimates of health effects from worst-case credible accidents can be made from present information. Nonetheless, dose commitments to the most exposed individual (80 rem/event) and to the population within 100 km of the site (4000 man-rem/ event) have been estimated for the on-pad accident (Bechtel 1979a). More work would be needed concerning the integrity of the nuclear waste container systems that would be employed for the space disposal option and the actual accident environments that would result. Additionally, the relationship between shielding and possible health effects during recovery from major accidents would require further technical study. Under accident conditions, however, the stability of the HLW is expected to reduce the consequences of any loss of containment (DOE 1979a).

In a space disposal reentry and burnup accident, the estimated average and individual dose is "quite small", yet the total population dose could be very large (e.g., about 10^7 man-rem/accident to the world population) (Bechtel 1979a).

Nonradiological Impacts. Generally, environmental impacts that would be caused by normal operations or nonradiological-type accidents from a space disposal option are not expected to be significant (NASA 1978). Potential environmental impacts related to the normal operations of space transportation systems that might be unique are discussed below.

The types of environmental health impacts that could be attributed to normal space transportation activities are:

- Gaseous and particulate emissions from rocket engines
- Noise generated during launches and landings (including sonic booms)
- Commitments of nonrecoverable resources.

These effects have been studied by NASA and an environmental impact statement has been issued (NASA 1978). To date, research has indicated there would be no significant effects to the human population from a steady launch rate of 60 shuttle flights per year.

During abnormal conditions, the major nonradiological concern appears to be whether or not large pieces of metal would reach the ground in the event of an upper stage failure. This question and others are the subject of ongoing investigations.

Natural System Impacts

Radiological and nonradiological impacts are analyzed below for the natural system.

Radiological Impacts. Environmental studies of the Barnwell Nuclear Fuel Plant (AGNS 1971, 1974; Darr and Murbach 1977) provide information concerning environmental impacts expected from normal processing of the reference waste mix. Expected environmental effects include modest heat additions to local water systems, as well as both gaseous and liquid releases of radioactive and nonradioactive materials.

In general, normal operation within regulatory limits should assure that ecosystem radiological impacts are acceptable. These conclusions are confirmed by generic studies (DOE 1979b).

The data base for environmental assessment of the space option is very preliminary at this time. Environmental assessments could be made only when the total system has been better defined. Bechtel (1979a) provides a recommended schedule for assessing ecosystem impacts from abnormal events, which, if adhered to, would make preliminary results available late in 1980.

Nonradiological Impacts. The major environmental impacts from construction of required waste treatment, payload fabrication, payload receiving, and launching facilities would be qualitatively similar to those of other construction activities. Construction impacts, in general, are related to resource commitments (land, water, and materials) and to effects on environmental quality and biotic communities from the pollutants and fugitive dust released by construction activities.

Water quality would be adversely affected by the creation of sedimentation resulting from runoff at construction sites, discharge of treated wastewaters and blowdown at reprocessing facilities, and salt pile runoff at the secondary waste repository (Bechtel 1979a).

Air quality during construction would be adversely affected as a result of fugitive dust and diesel equipment emissions, emissions from waste and employee transportation, and salt drift (Bechtel 1979a). On the basis of results of analyses performed for air quality, water quality, land quality, weather, and ecology during normal operations, no long-term or cumulative effects are predicted for the abiotic and biotic communities (NASA 1978).

Accidents related to Space Shuttle launches (without payloads) have been described elsewhere (NASA 1978) and are not expected to be environmentally significant.

Socioeconomic Impacts

Manpower estimates for construction and operation are a key variable in assessing socioeconomic impacts. Employment related to payload handling and launch is a differentiating factor between the space option and other waste disposal options.

Only preliminary data for the socioeconomic assessment of the space option are available at this time. A detailed assessment of the socioeconomic implications of the space disposal option would require more accurate employment estimates, information on the industrial sectors affected by capital expenditures, and identification of the specific geographic areas involved. Rochlin et al. (1976) provide a general discussion of the socioeconomic implications of nuclear waste disposal in space.

(a) While Kennedy Space Flight Center has already adjusted to many of the impacts mentioned below, selection of an alternative launch site would require additional impact assessment.

- Public Sector Economy. Current estimates of launch rates suggest that support of the entire space transportation system for the space disposal activity might require 25,000 to 75,000 employees. This work force represents a substantial payroll and a large number of households throughout the country that would constitute sizable demands for goods and services. The environmental impact statement for the Space Shuttle (NASA 1978) provides insight as to where money would be spent.
- Private Sector Economy. In addition to direct employment, the space disposal option would induce secondary employment, as well as major capital investment. This additional economic activity would, in turn, generate additional demands for goods and services.
- Population Size and Growth Rate/Population Composition. The size and geographic distribution of the work force levels would affect the magnitude and location of the socioeconomic impacts. The ability of local areas to meet such demands will affect the severity with which these impacts are perceived. Greater project definition and detail are necessary before these impacts can be accurately assessed.

Aesthetic Impacts

Aesthetic impacts for those aspects of the program unique to space disposal would be generally limited to noise and visual features.

Noise. Only the Orbiter reentry would produce sonic boom over populated areas. Extensive studies of sonic boom dynamics indicate that the maximum effects would be at the nuisance or annoyance level (NASA 1978).

Appearance. Visual effects are expected to be significant because of the eight-story preparation facility and a 100-m stack for the reprocessing facility. Of course, actual site selection could have a mitigating effect on these impacts (Bechtel 1979a).

Resource Consumption

Launches of space vehicles always commit certain resources that are never recovered.

Energy. Estimated total energy requirements for the space disposal program (construction plus 40-year operation), which are considered significant, are summarized below (Bechtel 1979a).

<u>Resource</u>	<u>Amount</u>
Propane, m ³	1.0 x 10 ⁷
Diesel fuel, m ³	1.5 x 10 ⁶
Gasoline, m ³	1.3 x 10 ⁵
Electricity, kWhr	5.9 x 10 ¹⁰
Propellants, MT	
Liquid hydrogen	2.7 x 10 ⁵
Liquid oxygen	3.7 x 10 ⁶
Rocket propellant	7.2 x 10 ⁵
Nitrogen tetroxide	2.4 x 10 ⁴
Monomethyl hydrazine	2.0 x 10 ⁴

Critical Resources. Estimated commitment of critical material resources required for construction plus 40 year operation (other than those required for launching) are characterized as follows (Bechtel 1979a).

<u>Resource</u>	<u>Amount</u>
Water, m ³	6.0 x 10 ⁷
Steel and Major Alloys, MT	
Carbon Steel	2.9 x 10 ⁵
Stainless Steel	3.0 x 10 ⁴
Chromium	5.0 x 10 ³
Nickel	2.0 x 10 ³
Major Nonferrous Metals MT	
Copper	3.8 x 10 ⁴
Lead	2.9 x 10 ³
Zinc	6.0 x 10 ²
Aluminum	8.3 x 10 ⁴
Concrete, m ³	1.1 x 10 ⁶
Lumber, m ³	4.0 x 10 ⁵

Land. Approximately 9000 ha (22,230 acres) of land would be required for the space disposal program. There is sufficient land capacity at the Kennedy Space Center to meet this requirement (Bechtel 1979a).

International and Domestic Legal and Institutional Considerations

The space disposal option has elements that are unique and that would have to be addressed in a comprehensive analysis of this alternative. For example, careful assignment of responsibility and accountability will have to be made among the federal agencies that would be involved in this disposal option.

The space disposal option would also present international concerns that would have to be recognized and addressed. Potential issues are:

- Risk of accidents affecting the citizens of countries that did not participate in the waste disposal decision
- Possibility of joint disposal programs with other countries
- Assignment of associated costs to various countries.

In addition to these generic international issues, there are a number of specific multi-national treaties, conventions, and agreements currently in force and subscribed to by the U.S. that bear upon the use of space for nuclear waste disposal. These include:

- "Treaty on Principles Governing the Activities of States in the Exploration and Use of Outer Space Including the Moon and Other Celestial Bodies" (1967)
- "Convention on International Liability for Damage Caused by Space Objects" (1972)
- "Agreement on the Rescue of Astronauts, the Return of Astronauts, and the Return of Objects Launched into Outer Space" (1972)
- "Convention on Damage Caused by Foreign Aircraft to Third Parties on the Surface" (1952)
- "Convention on Registration of Objects Launched into Outer Space" (1976).

This list suggests various issues that would have to be thoroughly explored in this early decision-making phase, including: (1) accident liability, (2) exclusive use of the lunar surface or other regions of outer space, and (3) international program involvement (e.g., use of the sea). These issues relate mainly to accident situations rather than routine operations.

In addition to these political and international issues, space disposal of nuclear waste would have a number of legal complexities associated with it, including liability and regulatory requirements (e.g., licensing). These concerns would be quite evident not only during, but also before and after actual implementation. Moreover, legal concerns could lengthen the time needed to implement a space disposal option.

6.1.8.5 Potential Impacts Over Long Term (Postemplacement)

Postemplacement for the space option is defined as the period of time after achievement of a stable solar orbit. Potential impacts during this period are analyzed for two different events: engineering failure and inadvertent human intrusion.

Potential Events

The possibility of sudden failure of a container in solar orbit would be extremely remote. However, if a container should rupture, for example, as a result of a meteor impact or degradation over the long term, the contents would be released and begin to spread. The physical processes by which the nuclear waste material would be dispersed in solar space include sputtering, thermal diffusion, and interactions with solar radiation and wind. Large pieces or particles of waste material would be sputtered into smaller particles, which in turn would disperse. The smallest particles, with radii less than 10^{-5} to 10^{-4} cm, would be swept out of the solar system by direct solar radiation pressure. Larger particles, those with radii up to 10^{-3} to 10^{-2} cm, would gradually lose momentum through scattering, charge exchange interactions, and collisions with energetic photons and solar wind protons. This process, called the Poynting Robertson effect, would cause these particles to begin moving in toward the sun where they would eventually be vaporized and broken down into smaller particles. Once this had occurred, the smaller particles would be swept out of the solar system by solar radiation pressure. This sweeping-out process would take an estimated 1000 to 10,000 years (Brandt 1970). NASA is currently studying this process.

The potential hazard from the isolated nuclear waste to persons on future space missions traversing the region about 0.85 A.U. is not known, but is believed to be extremely small and would be zero unless a manned trip by or to Venus were undertaken. Nuclear waste launched into an 0.85 A.U. orbit would not be recoverable for all practical purposes and the 0.85 A.U. solar orbit is far enough from the Earth and sufficiently stable that future Earth encounters would be effectively precluded (Friedlander et al. 1977).

Potential Impacts

With space disposal, waste would be isolated from the Earth for geologic time periods, in effect, permanently. Consequently, no long-term radiological or nonradiological health impacts are expected. The terrestrial component, storing only non-HLW, would therefore be minimized.

With regard to natural systems, upon retirement of waste processing fabrication and/or storage facilities (including the payload preparation facility), the land areas could be returned to other productive uses. Although details of decommissioning are not available, the various alternatives should not have a significant effect on the program. Beneficial uses of the sites by future generations would not be hindered.

6.1.8.6 Cost Analysis

Space disposal costs can be identified as follows (Bechtel 1979a):

- Waste processing/encapsulation (this may be incremental for comparisons with other alternatives)
- Ground transportation
- Launch facilities and space hardware (reusable and expendable)
- Launch operations and decommissioning
- Geologic disposal of residual nuclear wastes.

Although many of the basic space and waste technologies are understood, extrapolation to meet the requirements of the space disposal mission does not permit a valid cost estimate at this conceptual stage of the program. Initial scoping studies indicate that costs for many of these portions of the space disposal system would be similar to costs for other alternatives. The major cost difference for the space disposal alternative is attributable to the Space Shuttle operations. Capital, operating, and decommissioning costs for this incremental portion of the program are discussed briefly below.

Capital Costs

Capital costs would be incurred at Kennedy Space Center for construction of equipment dedicated to the waste disposal mission. This would include the special purpose transporter, launch pad; launch platform, and firing room. If these capital costs were recovered as

charges to DOE as a Space Shuttle user, as is contemplated for other Space Shuttle applications, they would accrue as operating costs to any DOE space disposal program. Therefore, these costs would be integrated in the per-flight charges under operating costs. One special facility not usable for other shuttle operations would be the payload preparation facility. Current estimates for this facility are \$29 million (1978 dollars). Other capital costs might accrue because of the need to allow radiation to decay in the HLW for at least 10 years prior to space disposal. Costs for such interim storage facilities have not been identified at this time.

Operating Costs

Operating costs for the space disposal alternative would be calculated on a per-flight basis, as they are for other participants in the Space Shuttle program. The per-flight cost would be approximately \$39 million in 1978 dollars.

The breakout of this estimate is:

- Up-rated Space Shuttle - \$16 million
- Orbit transfer vehicle - \$1.6 million
- Solar orbit insertion stage - \$1.6 million
- Reentry vehicles - \$5 million.

Decommissioning Costs

Decommissioning costs associated with Space Shuttle waste disposal operations would probably be limited to the facilities for waste processing and packaging, the only facilities at which contamination might be anticipated. Those decommissioning costs have been estimated at 10 percent of the initial capital costs; i.e., approximately \$3 million. Costs for decommissioning other facilities associated with the space disposal alternative are assumed to be similar to those for decommissioning facilities associated with other waste disposal alternatives.

6.1.8.7 Safeguard Requirements

Safeguards would be considered for both space disposal and the associated terrestrial disposal. For space disposal of HLW, the risk of diversion would be short-term. Once the waste had been successfully disposed of in accordance with the design, the probability of an unauthorized retrieval would be very low. Physical protection of the sensitive facilities and transportation operations would be the most effective way to deny access for the short term. Note that if this alternative were chosen for the once-through fuel cycle, despite the very high throughput required, on a purely safeguards basis it would compare favorably with many other alternatives because of the difficulty of retrieving material once it is successfully deployed. See Section 4.10 for further details on safeguards for applicable predisposal operations.

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6.2 COMPARISON OF ALTERNATIVE WASTE DISPOSAL CONCEPTS

This section provides an assessment of the nine waste management concepts discussed in Chapter 5 and Section 6.1 of this Statement.

For the reader's convenience, a brief review of each of the alternative concepts is first presented in Section 6.2.1. Next, ten assessment factors and a set of related standards of judgement are introduced. The first stage of the analysis follows, in which the concepts are screened using the standards of judgement introduced in the previous section. Concepts which remain after the screening are then compared on the basis of the assessment factors and most promising concepts identified.

6.2.1 Summary Description of Alternative Waste Disposal Concepts

This section presents brief descriptions of the nine waste management concepts considered in this comparison. Characteristics of each concept are described in more detail in Chapters 4 and 5 and Section 6.1. Technical approaches not summarized here have been advanced for certain concepts that if implemented might result in a waste management system differing from that described here. In addition, the developmental process might result in a system different than described here, especially for concepts currently in a very preliminary stage of development.

6.2.1.1 Mined Repository

In the mined repository concept, disposal of waste would be achieved by manned emplacement in mined chambers in stable geologic formations. Engineered containment would be provided by the waste form, canisters, overpacks, and sleeves. Use of a tailored backfill would provide an additional engineered barrier. Isolation and natural barriers would be provided by the host rock and surrounding geologic environment, which would be selected to provide stability, minimal hydrologic transport potential and low resource attractiveness.

A waste packaging facility would be located at the repository site where spent fuel assemblies would be individually sealed into canisters. The canisters would be incorporated into the multibarrier package and then would be placed in individual boreholes in the floor and walls of mined chambers 500 to 1,000 m deep in suitable host-rock formations. Backfill would be placed around each package following emplacement. As each chamber is ready, it would be backfilled with rock and sealed. When the repository is filled the access tunnels and shafts would be filled with appropriate materials and sealed.

All waste types referenced in Table 6.2.1 could be emplaced in the mined repository.

A reprocessing fuel cycle would produce high-level liquid waste that could be solidified to a stable waste form, packaged in canisters that are part of a multibarrier package, and emplaced in the mined repository. Transuranic waste^(a) would also be packaged and emplaced in the mined repository.

(a) Hulls, hardware, remotely handled and contact-handled TRU waste. See Table 6.2.1.

TABLE 6.2.1. Disposition of Principal Waste Products Using the Proposed Waste Disposal Concepts

	<u>Spent Fuel Assemblies</u>	<u>High-Level Liquid (Fuel Processing Waste)</u>	<u>TRU Waste(a)</u>
Mined Repository	Packaged and emplaced in mined repository.	Incorporated in immobilized solid, packaged and emplaced.	Packaged and emplaced in mined repository.
Very Deep Hole	Packaged and emplaced in deep hole repository.	Converted to immobilized solid. Packaged and emplaced in deep hole repository.	Disposal using suitable alternative technique.(b)
Rock Melt	Processed to a liquid state	Poured in rock melt repository.	Disposal using suitable alternative technique.(c)
Island Mined Repository	Packaged and emplaced in island mined repository.	Converted to immobile solid. Packaged and emplaced in island repository.	Packaged and emplaced in island mined repository.
Subseabed	Packaged and emplaced in subseabed repository.	Converted to immobile solid. Packaged and emplaced in subseabed repository.	Disposal using suitable alternative technique.(b)
Ice Sheet	Packaged and emplaced in ice sheet repository.	Converted to immobile solid. Packaged and emplaced in ice sheet repository.	Disposal using suitable alternative technique.(b)
Well Injection	Processed	Injected into geologic formations.	Disposed using suitable alternative concept.
Transmutation	Processed	Selected isotopes partitioned and transmuted to stable or shorter lived isotopes and disposed of using alternative concept.	Disposed using suitable alternative concept.
Space	Processed	Entire waste stream or selected isotopes converted to solid and emplaced in heliocentric orbit.	Disposed using suitable alternative concept.

(a) Remotely handled and contact-handled TRU wastes including dissolver solids, HEPA filters, incinerator ash wastes, failed and decommissioned equipment wastes.

(b) Could possibly be disposed of by the concept, but this is considered unlikely.

(c) Some chopped cladding and TRU wastes might be slurried into rock melt cavity subject to diluting limitations on HLW waste.

6.2.1.2 Very Deep Hole

In the very deep hole concept, disposal of high-level waste would be achieved by remote emplacement in bored shafts at depths greatly exceeding those of the mined repository. Engineered containment would be provided by the waste form, canisters, and perhaps additional barrier layers. Sorptive backfill, if used, would provide an additional engineered barrier. Isolation and natural barriers would be provided by the host rock and surrounding geologic and hydrologic environment, enhanced by the great distance to the accessible environment. The geologic and hydrologic environment would be selected to provide stability, minimal hydrologic transport potential, and low resource attractiveness.

A waste packaging facility would be located at the repository site where spent fuel assemblies would be packaged individually. The packaged fuel assemblies would be placed in rotary drilled holes as much as 10,000 m deep in crystalline rock. Holes for packages for

fuel assemblies would be approximately 48 cm in diameter. After emplacement of approximately 150 packages in the bottom 1,500 m of the hole, the hole would be sealed and filled.

A reprocessing fuel cycle would require that prior to emplacement, high-level liquid waste be converted to an immobile solid and incorporated into a multibarrier package compatible with the very-deep hole environment. TRU waste resulting from reprocessing would be disposed using other suitable disposal concepts (Table 6.2.1).

6.2.1.3 Rock Melting

In the rock melting concept, disposal of high-level and some TRU waste would be achieved by remote emplacement of liquid or slurried waste into a mined cavity. Decay heat would be allowed to melt the surrounding rock which eventually would solidify, and form a solid, relatively insoluble, rock-waste matrix. Engineered containment could be provided during the operational period by a temporary chamber lining; however, engineered barriers would not be present during the molten phase. Following solidification, the rock-waste matrix would provide quasi-engineered containment wherein the host rock and waste forms would provide suitable post-solidification properties. Isolation and natural barriers would be provided by the surrounding geologic and hydrologic environment which would be selected to provide stability, minimal hydrologic transport potential and low resource attractiveness.

Spent fuel would be converted to a slurry or dissolved at a waste processing facility located at the repository site. Plutonium and uranium could be chemically separated and sent to a mixed oxide fuel fabrication facility if a reprocessing fuel cycle were utilized. High-level waste and contact-handled TRU waste in liquid or slurry form would be piped separately to the repository. Here the waste would be injected into mined cavities approximately 20 m in diameter and 2,000 m deep. Liquid or slurried contact-handled TRU waste, supplemented with water as required, could be injected into the cavity to provide cooling. After the cavity is filled, cooling would be terminated and the injection shaft sealed. Heat from radioactive decay would melt the surrounding rock, forming a molten rock-waste mix at a temperature $\geq 1000^{\circ}\text{C}$. The mix would eventually solidify, trapping the waste within a rock matrix. Solidification should be complete in about 1,000 years.

Fuel hardware and TRU waste for which conversion to liquid or slurry is impractical would be packaged and emplaced using a suitable alternative disposal concept (Table 6.2.1).

6.2.1.4 Island Mined Repository

In the island mined repository concept, disposal of waste would be achieved by manned emplacement in mined chambers in stable geologic formations on continental islands. Engineered containment would be provided by the waste form and multibarrier package. Tailored sorptive backfill would provide an additional engineered barrier. Isolation and natural barriers would be provided by the host rock and the surrounding geologic and hydrologic environment which would be selected to provide stability, minimal hydrologic transport potential and low resource attractiveness.

Spent fuel assemblies would be packaged individually into canisters at a waste packaging facility located in the continental U.S. All canisters would be loaded into shipping casks and transported by rail to the embarkation port facility. At the port facility the waste packages would be transferred from the rail casks to ocean shipping casks which would be loaded aboard ocean-going vessels. These vessels would transport the waste to a receiving port on the U.S.-owned repository island. Waste casks would be transferred to rail or highway vehicles for shipment to the repository site. Here the canisters would be unloaded from the shipping casks, placed in multibarrier packages, and placed in individual boreholes in the floor of mined chambers at least 500 m deep in granite or basalt, located either within the fresh groundwater lens or within underlying saline groundwater. Backfill would be placed around each package following emplacement. As each chamber is ready it would be backfilled and sealed. When the repository is filled the access tunnels and shafts would be backfilled with appropriate materials and sealed.

A reprocessing fuel cycle would require high-level liquid waste to be converted into an immobile solid that would be incorporated into a multibarrier package compatible with the island geologic environment. Other wastes would be packaged and emplaced in the island repository.

6.2.1.5 Subseabed Disposal

In the subseabed disposal concept, disposal of waste would be achieved by remote emplacement in relatively thick, stable beds of sediment located in deep, quiescent, and remote regions of the oceans. Engineered multibarrier containment would be provided by the waste form, canister, and the outer body of the emplacement container. Isolation and a natural barrier would be provided by clay sediments which would be chosen for uniformity, high plasticity, low permeability, high sorption potential, long-term stability and low resource attractiveness. The ocean itself would enhance remoteness, providing protection from human intrusion. Because the ocean is part of the accessible environment it would not be considered as a barrier to waste release.

Spent fuel assemblies would be packaged individually in canisters at a waste packaging facility located in the continental U.S. Packaged fuel assemblies would be loaded into shipping casks and transported by rail to the embarkation port facility. At the port facility waste packages would be removed from the shipping casks and loaded into emplacement vehicles, probably free fall penetrometers. These would be loaded onto special oceangoing vessels and transported to the emplacement site, located in the mid-plate, mid-gyre region of the ocean with depths of 3,000 to 5,000 m. At the site the penetrometers would be released to penetrate 50 to 100 m into the clay sediment. Closing of the hole above the penetrometers might occur spontaneously or be accomplished by mechanical means and would seal the waste into the sediment. A monitoring vessel would verify satisfactory emplacement.

A reprocessing fuel cycle would produce liquid high-level waste that would be converted to an immobile solid for incorporation into a multibarrier package designed for emplacement in the sediments. TRU waste would probably require another suitable disposal concept (Table 6.2.1).

6.2.1.6 Ice Sheet Disposal

In the ice sheet disposal concept, disposal of high-level waste would be achieved by remote emplacement within a continental ice sheet. The plasticity of the ice would eventually seal the waste from the environment and subfreezing temperatures would preclude hydrologic transport except possibly at the conditions encountered at the ice-rock interface. Engineered multibarrier containment would be provided by the waste form and canisters and possibly overpacks. Isolation and a natural barrier would be provided by the ice mass. The geographic location of the repository and the inclement weather of continental ice sheets would contribute to the remoteness of the repository and decrease the possibility of human intrusion.

Spent fuel assemblies would be packaged individually in canisters at a waste processing facility located in the continental U.S. Packaged fuel assemblies would be loaded into shipping casks and transported by rail to the embarkation port facility. At the port facility waste packages would be transferred from rail casks to ocean-shipping casks which would be loaded aboard ocean-going vessels. These vessels would transport the waste to a receiving port at the ice margin. Here the waste packages in shipping casks, would be transferred to tracked vehicles for transport to the repository, located some distance inland. At the repository site the waste packages would be removed from the transport casks, placed into pilot holes drilled 50 to 100 m into the ice and tethered to anchor plates with 200 to 500 m cables or allowed to melt freely into the ice. Heat from radioactive decay would melt the ice and the package would sink into the ice sheet, reaching its final position in six to eighteen months. The pilot holes would be sealed by filling with water which would subsequently freeze. Refreezing of water above the package as it progressed downward would complete sealing of the emplacement holes.

A reprocessing fuel cycle would produce liquid high-level waste that would be converted to an immobile solid compatible with the ice environment. This solidified waste would be packaged and emplaced in the ice sheet repository. TRU waste would probably be disposed using an alternative disposal concept (Table 6.2.1).

6.2.1.7 Well Injection

In the well injection disposal concept, disposal of high-level waste would be achieved by remote emplacement of liquid or slurried waste into stable geologic formations capped by an impermeable boundary layer. A degree of engineered containment would be supplied by the waste form if a grout were used but would not be present during the injection phase. Isolation and natural barriers would be provided by the host rock and the surrounding geologic and hydrologic environment which would be selected for its stability, minimum hydrologic transport potential, high sorption potential and low resource attractiveness.

A waste processing facility would be located at the repository site where spent fuel would be dissolved and prepared for injection, either directly as a dilute acidic liquid or as a neutralized grout. The prepared waste would be transferred by piping to the injection well field. Dilute acid waste, if used, would be injected into porous sandstone having shale caprock at depths of approximately 1,000 m. Neutralized grout would be injected into a shale formation having natural or induced fractures at depths of approximately 500 m. TRU waste would require an alternative disposal concept.

Liquid high-level waste resulting from a reprocessing fuel cycle would be transferred directly to the waste preparation facility, colocated with the reprocessing plant. TRU waste would be packaged and emplaced using an alternative disposal concept (Table 6.2.1).

6.2.1.8 Transmutation

Transmutation would function as an ancillary waste treatment process for the conversion of selected long-lived waste isotopes to shorter-lived isotopes potentially reducing the time during which repository integrity must be maintained. The process would be operated in conjunction with a waste management system using a suitable alternative disposal concept for disposal of radioactive waste, including transmutation products (Table 6.2.1). Because transmutation is a waste treatment process and not a disposal alternative, it cannot be assessed in terms of containment, barriers and remoteness in the same manner as these terms are applied to repositories.

At a processing plant spent fuel would be dissolved and uranium and plutonium separated for recycle. Reprocessing wastes would be transferred to an adjacent partitioning facility where long-lived waste isotopes would be partitioned from the reprocessing waste stream. The residual waste streams, stripped of long-lived isotopes, would be processed for disposal using a suitable disposal concept. The isotopes selected for transmutation would be combined with recovered plutonium and uranium and shipped to a MOX-FFP.

At the fuel fabrication plant the plutonium-uranium-waste isotope mixture would be fabricated into MOX fuel assemblies following addition of sufficient enriched uranium to achieve the desired end-of-cycle reactivity. TRU waste from the fuel fabrication plant would be sent to a colocated waste purification facility for recovery of waste actinides. Recovered actinides would be returned to the fuel fabrication facility for incorporation into MOX fuel; the residual waste would be processed for disposal using a suitable alternative waste disposal concept (Table 6.2.1).

The MOX fuel, containing the waste isotopes for transmutation, would be shipped in shielded casks to power reactors where a portion of the waste isotopes would be transmuted to stable or shorter-lived isotopes. Transmuted isotopes would be partitioned for disposal during the subsequent reprocessing cycle. Repeated cycles would be required to achieve nearly complete transmutation of the long-lived isotopes.

Implementation of transmutation as an actinide waste treatment process requires that spent fuel be reprocessed to recover the actinides and that the actinides be recycled for transmutation, mandating a reprocessing-type fuel cycle.

6.2.1.9 Space

In the space disposal concept, disposal of selected waste products would be achieved by insertion of waste packages into a stable solar orbit approximately half-way between the orbits of Earth and Venus. Engineered containment would be provided by the waste form and its engineered package. Isolation would be provided by the remoteness of the orbit from Earth and the stability of the orbit. An additional impediment to return of waste would be provided by inclining the orbit to the ecliptic.

Spent fuel would be chopped and dissolved at a processing facility. Plutonium and uranium would be chemically separated and sent to a MOX-FFP if a reprocessing fuel cycle were utilized. Waste products for which space disposal is intended would be partitioned from the waste stream and transferred to an adjacent waste preparation facility. High-level and contact-handled TRU waste not destined for space disposal would be processed for disposal using a suitable alternative disposal concept (Table 6.2.1). Alternatively, the entire liquid high-level waste stream, including uranium and plutonium constituents, could be transferred to the waste preparation facility for space disposal.

At the waste preparation facility, the waste would be incorporated into a solid ceramic-metal composite ("cermet") which would be formed into a payload of suitable shape and size. The payload would be packed into a radiation shield and this assembly loaded into a shipping cask for transport to the nuclear payload preparation facility near the launch site.

At the nuclear payload preparation facility, the shielded waste assembly would be removed from the shipping cask and loaded into a reentry vehicle. A special transporter would then take the assembly to the launch site, where it would be positioned in the space shuttle cargo bay with an orbit transfer vehicle and a solar orbit insertion stage.

The space shuttle would be launched into earth orbit where the reentry vehicle-payload assembly would be deployed from the cargo bay. The shielded waste assembly would then be removed from the reentry vehicle and attached to the solar orbit insertion stage, which would be latched to the orbit transfer vehicle. The orbit transfer vehicle would propel the solar orbit insertion stage into an earth escape trajectory, release the solar orbit insertion stage and return to earth orbit for recovery. The solar orbit insertion stage and the waste would continue and the waste would ultimately be inserted into a stable solar orbit at 0.85 astronomical units. The space shuttle would return to earth carrying the reentry and orbit transfer vehicles.

6.2.1.10 Summary

The relationships of the nine disposal concepts to the waste products of the two primary fuel cycles have been summarized in Table 6.2.1. Products of the once-through fuel cycle include spent fuel assemblies with probably a small stream of contact-handled TRU waste resulting from fuel element failures. Five of the disposal concepts could dispose of these products directly. However, rock melt, well injection, transmutation and space disposal would require processing the spent fuel to liquid or slurry form with the result that

the spectrum of waste products characteristic of the reprocessing fuel cycle is generated. This includes liquid high-level waste, fuel hulls and hardware, and a substantial quantity of remotely handled and contact-handled TRU waste. It should be noted that the reprocessing fuel cycle will likely require an alternative disposal facility (probably a mined repository) for the high volume TRU wastes for all concepts except the island repository; mined repositories; and, perhaps, the subseabed.

6.2.2 Assessment Factors and Standards of Judgement

Ten assessment factors have been selected to facilitate comparison of the proposed waste management concepts. These factors are discussed in Subsections 6.2.2.1 through 6.2.2.10. Associated with certain of these factors are standards of judgement. The standards of judgement are applied in Section 6.2.3 to reduce the nine proposed waste management concepts to a subset of candidate concepts with greatest potential for adequate performance. Concepts in this subset are then compared in Section 6.2.4 on the basis of the ten assessment factors. The ten assessment factors are listed in Table 6.2.2 below; the assessment factors are underlined. The standards of judgement appear as bullets in Table 6.2.3 and are grouped under the (underlined) assessment factors.

TABLE 6.2.2. Assessment Factors

Radiological Effects

- operational period
- post-operational period

Non-Radiological Environmental Effects

- health effects
- socio-economic effects
- aesthetic effects
- ecosystem effects

Current Status of Development

- availability of technology
- availability of performance assessment methodologies

Conformance with Federal Law and International Agreements

Independence from Future Development of the Nuclear Industry

- industry size
- fuel cycles
- reactor design

Cost of Development and Operation

Potential for Corrective or Mitigating Action

Long-Term Maintenance and Surveillance Requirements

Resource Consumption

Equity of Risk

TABLE 6.2.3. Standards of JudgementRadiological Effects

- A concept should comply with radiological standards established for other fuel cycle facilities.
- Containment should be maintained during the period dominated by fission product decay.
- Waste should be isolated from the accessible environment for a minimum of 10,000 years.

Non-Radiological Environmental Effects

No standards were advanced for this factor.

Current Status of Development

- The concept should be amenable to development within a reasonable period of time such that implementation is not left to future generations.
- Implementation of a concept should not require scientific breakthroughs.
- Capabilities for assessing the performance of a concept must be available prior to committing major R&D programs to its development.

Conformance with Federal Law and International Agreements

No standards were advanced for this factor.

Independence from Future Development of the Nuclear Industry

- Implementation of a concept should not be dependent upon the size of the nuclear industry.
- Concepts should be independent of fuel cycle issues.
- Concepts should be independent of reactor design issues.

Cost of Development and Operation

No standards were advanced for this factor.

Potential for Corrective or Mitigating Action

- Concepts should allow corrective action to be taken in case of failure of a system to perform as designed.

Long-Term Maintenance and Surveillance Requirements

- Reliance should not be placed on maintenance or surveillance for extended times following termination of the operational period.

Resource Consumption

No standards were advanced for this factor.

Equity of Risk

No standards were advanced for this factor.

6.2.2.1 Radiological Effects

A central objective of the nuclear waste management program is to limit radiation dose to both the public and to operating personnel to acceptably low levels. Two time periods are of interest. One is the operational period involving waste treatment, transportation, and emplacement and the second is the post operational period following termination of repository operations.

A useful measure of radiological effects during the operational period is radiation exposure resulting from emplacement of a quantity of waste derived from the generation of a

unit of electrical power by nuclear means. Unfortunately, the current state of development of many of the concepts does not permit computation of this measure. Therefore, this analysis will rely upon relative comparison, using processing and transportation requirements as secondary indicators of potential radiation dose during the operational period.

A reasonable minimum level of radiological performance during the operating period is that risks shall not be greater than those allowed for other nuclear fuel cycle facilities. This suggests a standard that appropriate regulatory requirements established for other fuel cycle facilities be met.

Objectives 1 and 2 of the proposed DOE Waste Management Performance Objectives (Table 6.2.4) are intended to provide standards related to the radiological performance of waste management concepts during the post-emplacement period. Objective 1 requires that waste containment within the immediate vicinity of initial placement should be virtually complete during the period when radiation and thermal output are dominated by fission product decay. Objective 2 requires a standard of reasonable assurance that wastes will be isolated from the environment for a period of at least 10,000 years with no prediction of significant decrease beyond that time. Both standards were adopted for this analysis (Table 6.2.3).

TABLE 6.2.4. Proposed DOE Waste Management Performance Objectives(a)

1. Waste containment within the immediate vicinity of initial placement should be virtually complete during the period when radiation and thermal output are dominated by fission product decay. Any loss of containment should be a gradual process which results in very small fractional waste inventory release rates extending over very long release times, i.e., catastrophic losses of containment should not occur.
2. Disposal systems should provide reasonable assurance that wastes will be isolated from the accessible environment for a period of at least 10,000 years with no prediction of significant decreases in isolation beyond that time.
3. Risks during the operating phase of waste disposal systems should not be greater than those allowed for other nuclear fuel cycle facilities. Appropriate regulatory requirements established for other fuel cycle facilities of a like nature should be met.
4. The environmental impacts associated with waste disposal systems should be mitigated to the extent reasonably achievable.
5. The waste disposal system design and the analytical methods used to develop and demonstrate system effectiveness should be sufficiently conservative to compensate for residual design, operational, and long-term predictive uncertainties of potential importance to system effectiveness, and should provide reasonable assurance that regulatory standards will be met.
6. Waste disposal systems selected for implementation should be based upon a level of technology that can be implemented within a reasonable period of time, should not depend upon scientific breakthroughs, should be able to be assessed with current capabilities, and should not require active maintenance or surveillance for unreasonable times into the future.
7. Waste disposal concepts selected for implementation should be independent of the size of the nuclear industry and of the resolution of specific fuel cycle or reactor design issues and should be compatible with national policies.

(a) DOE/NE-0007--Statement of Position of the United States Department of Energy in the Matter of Proposed Rulemaking on the Storage and Disposal of Nuclear Waste.

Non-Radiological Environmental Effects

Non-radiological environmental effects considered to be of potential significance in the comparison of waste management concepts include health effects from non-radiological causes, socioeconomic effects, aesthetic effects, and effects on ecosystems.

Health effects from non-radiological causes include injuries and deaths occurring to both occupational workers and to the general public from routine operations and from accidental conditions.

Socioeconomic effects include impacts on the well-being of communities in the vicinity of waste management facilities.

Potential aesthetic effects include noise, odor and impacts on visual resources.

Both natural and managed ecosystems would be affected by waste management operations. Potential impacts include those on ecosystem productivity, stability, and diversity.

No standards of judgement have been advanced for non-radiological environmental effects, although all concepts would be expected to comply with standards established by responsible Federal and state regulatory agencies. The proposed DOE Performance Objective 4 asserts the importance of minimizing non-radiation-related environmental effects.

6.2.2.2 Status of Development

This factor is intended to assess the waste management concepts on the basis of the maturity of the concepts. Two issues are of concern: 1) availability of technology required to implement the concept, including that required for site characterization, repository development, waste treatment, handling, emplacement, and monitoring; and, 2) ability to predict performance of the waste management system. A third issue, cost of research and development, is considered under the factor of cost.

Three standards of judgement relating to status of development can be derived from the proposed DOE Performance Objective 6. First the technology must be implemented within a reasonable period of time where "reasonable period of time" implies that those currently responsible can complete the major part of implementing a concept and not pass an unresolved problem on to future generations. Consequently, Objective 6 also states that scientific breakthroughs should not be required to permit implementation of a concept. Further capabilities for assessing the performance of any particular waste management concept must be available at the time that a decision is made to place emphasis on the development of any particular concept.

6.2.2.3 Conformance with Federal Law and International Agreements

The purpose of this factor is to identify and compare potential conflicts with Federal legislation and international treaties, conventions, and understandings to which this nation is a party that would prevent implementation of a proposed option. The DOE proposed Performance Objective 7 states that waste management systems "should be compatible with national

policies" suggesting that concepts might be rejected because of potential policy conflicts. Because Federal legislation and international agreements can be amended for reasonable cause, this condition will not be used as a standard, but its consideration provides insight into the difficulty of implementation. Any waste management concept, if implemented, would be required to comply with applicable laws and regulations.

6.2.2.4 Independence from Future Development of the Nuclear Industry

Implementing a nuclear waste management system is a large scale, costly, and long-term effort. Concepts selected for priority development should be independent of the future development of the nuclear industry including industry size, fuel cycles, and reactor designs.

Three standards of judgement derived from DOE Performance Objective 7 are related to this factor: 1) waste disposal concepts selected for implementation should be independent of the size of the nuclear industry, 2) independent of specific fuel cycles and 3) independent of reactor design issues.

6.2.2.5 Cost of Development and Operation

The purpose of this factor is to compare concepts on the basis of estimated costs for research and development (presumably to be borne by the Federal government but recovered from the utilities through fees charged for disposal) and on costs of implementation and operation (borne by utilities and included in their rate bases). No standards have been established for cost.

6.2.2.6 Potential for Corrective Action

The probability of system failure can be reduced to low levels by careful design, thorough assessment of performance and provision of redundant systems. However, as with any engineered system, probability of failure cannot be entirely eliminated, with the result that there will remain a probability (although very low) that the system may not perform as expected. Thus the ability to detect and correct failure or to mitigate its consequences would be a desirable property of the concept selected for implementation. The desirability of corrective action capability is implied by DOE Performance Objective 5 which suggests that corrective action capabilities should be provided to compensate for residual uncertainties in system performance. Thus the importance of corrective action capability should be assessed with consideration of residual uncertainties in system performance.

The proposed NRC Technical Standards for Regulating Geologic Disposal of High-Level Radioactive Waste require retrievability, a form of corrective action, to be maintained for 50 years following termination of waste emplacement operations (Proposed 10 CFR 60.111(a)(3)). No standards were established for corrective action potential given the dissimilar characteristics of certain of the waste management options.

6.2.2.7 Long-Term Maintenance and Surveillance Requirements

Future generations cannot reasonably be expected to assume a burden of maintaining and monitoring the nuclear wastes of present generations. Thus a desirable assessment factor for waste management concepts is that they require minimal maintenance or monitoring following decommissioning. The Environmental Protection Agency has included in its draft standards for waste management a stipulation that surveillance and maintenance should not be relied upon for a period exceeding 100 years after termination of active disposal operations (43 Fed. Register, Section 221, November 1978). A more general performance standard was adopted for this analysis that reliance should not be placed on maintenance and surveillance for extended times following termination of the operational period.

6.2.2.8 Resource Consumption

Any waste management option would require the consumption of certain resources including energy, critical nonfuel materials, and land. Certain materials which are important to a waste management option may be in short supply, potentially producing market disruptions or increased dependence on uncertain supplies. Potentially critical materials are listed in Table 6.2.5. It is important that no waste isolation approach use an unreasonable amount of any critical resource, but no specific standard is advanced.

TABLE 6.2.5. Potentially Critical Materials(a)

Aluminum	Cobalt	Nickel	Water
Antimony	Columbium	Platinum	Natural Gas
Asbestos	Graphite	Potash	Electricity
Bismuth	Iodine	Quartz (crystals)	Coal
Cesium	Manganese	Tantalum	Petroleum-Derived Fuels
Chromium	Mica	Tin	Other Fuels

(a) The nonfuel minerals of this group are considered to be "major problems from the national viewpoint" by the U.S. Bureau of Mines because of U.S. low-grade resource or reserve inadequacy to Year 2000

6.2.2.9 Equity of Risk

Although the responsibility for disposal of high level radioactive waste belongs to the Federal government, the implementation of a specific solution will require cooperation with the state and local governments, and with the general public. A few localities will be required to accept and service the facilities for disposal of waste that was created in providing service and benefits to a very broad segment of the country's population. Consequently, the implementation of a disposal method will have to be judged against the equity of risk by the political subdivision involved.

6.2.3 Application of Performance Standards

The nine proposed waste disposal concepts are examined in this section with respect to the performance standards advanced in Table 6.2.3. Results of this judgement are tabulated in Table 6.2.6. The subset of concepts meeting these standards are subjected to more detailed comparative analysis in Section 6.2.4.

6.2.3.1 A Concept Should Comply with Radiological Standards Established for Other Fuel Cycle Facilities

The unique characteristics of several of the proposed waste disposal concepts set them quite apart in design and operation from any existing fuel cycle facility. Thus, although it is appropriate to evaluate the concepts on current dose, risk and emission standards, it may be inappropriate to apply regulations relating to the means of achieving these standards. It is not evident, based on available information, that any of the nine proposed concepts would necessarily fail to comply with dose, risk and emission standards; though it is likely that the radiological releases would vary among the concepts.

6.2.3.2 Containment Should be Maintained During the Period Dominated by Fission Product Decay

"Containment" is defined in the NRC proposed technical criteria for regulating geologic disposal of high-level radioactive waste as "keeping radioactive waste within a designated boundary" (Proposed 10 CFR Part 60). Because of inherent differences among the concepts, the following definitions of containment are used for this assessment:

- Mined Repository--Waste is contained within the waste package (Proposed 10 CFR Part 60.)
- Very Deep Hole
Island Mined Repository
Ice Sheet Disposal } Waste is contained within the package.
- Rock Melt--Waste is contained within the rock-waste matrix, and in the intended location.
- Subseabed Disposal--Waste is contained within the package (penetrometer case or overpack).
- Well Injection--Dilute Acid: Waste is contained within the intended region of the host formation
Shale-Grout: Waste is contained within the grout matrix, and in the intended region of the host formation.
- Transmutation--None, the containment concept is not applicable.
- Space--Waste is contained within its package within the predetermined heliocentric orbit.

Based on these definitions of containment, engineering judgement indicates that containment for several hundred years could likely be achieved using the mined repository, very

deep hole, island mined repository, subseabed, ice sheet, and space disposal concepts. Uncertainties, however, are associated with the very deep hole concept depending on depth of emplacement and associated conditions of temperature and pressure to which the package is exposed.

Because the rock melt concept does not provide a system of engineered barriers, and because of the elevated temperatures, it appears likely that heated water vapor or liquid could contact, leach and transport waste from the as yet unsolidified rock-waste matrix of the rock melt concept during the initial 1000-year post-operational period.

Because the well injection concept does not provide a series of engineered barriers, one thousand year containment could not be assured with either of the well injection proposals. Diffusion of dilute acid injected waste into fractures and discontinuities of formations adjacent to the host formation could be expected.

In conclusion, it appears probable that containment of emplaced waste, as defined, could be maintained through the period dominated by fission product decay for all concepts except rock melt and well injection. The containment concept does not apply to transmutation.

6.2.3.3 Waste Should Be Isolated from the Accessible Environment for a Minimum of 10,000 Years

Ten thousand years has been proposed as a time period during which the radiotoxicity of properly treated waste would decay to levels comparable with the natural uranium ore bodies from which the materials were originally derived (Voss 1980). "Isolated" is interpreted as "segregation of the waste from the accessible environment within acceptable limits" (Proposed 10 CFR Part 60) where the accessible environment includes the atmosphere, the land surface, surface waters, oceans and presently used aquifers (Proposed, 10 CFR Part 60, 40 CFR Part 146). "Acceptable limits" has been generally interpreted to include releases resulting in dose rates within the normal variation of naturally occurring radiation dose rates (DOE 1980).

Analysis to date of the mined repository concept suggests no reason to believe that acceptable isolation could not be maintained by the geologic environment for a 10,000-year period, with the possible exception of very low probability catastrophic accident situations. The probability of these occurring is estimated to be small. Similarly, it appears quite possible that the very deep hole concept could maintain acceptable waste isolation over the required period if such depths are successfully isolated from ground water.

Maintenance of waste package containment cannot be assumed for the 10,000-year period for the mined repository, very deep hole, island mined repository, subseabed disposal and ice sheet disposal concepts. Package failure would expose the waste form to a saturated hydrologic environment for the subseabed and island disposal concepts and acceptable isolation would be dependent upon stability of the hydrologic environment and the sorptive properties of the host material and surrounding geologic environment. Available evidence indicates that acceptable isolation could be maintained using the subseabed concept. Satis-

factory performance of the island concept, while possible, is less certain because of an incomplete understanding of island hydrologic systems.

Maintenance of isolation for the requisite period under ice sheet conditions appears to be sufficiently questionable as to preclude this option from further consideration on the basis of this standard of judgement. If not tethered, the packages would descend to the ice-rock interface where the waste form packages could be pulverized by ice motion, and waste subsequently transported to the ocean by water potentially present at the interface. If tethered, ice sheet erosion or sublimation (possible within a 10,000-year period given historical climatic fluctuation) could expose waste to the surface environment.

The waste-rock matrix of the rock melt concept would potentially be exposed to severe hydrothermal alteration and leaching conditions late in the cooling phase when hot water may be present at the periphery of the rock-waste mass. This could result in transfer of waste to ground water. However if the surrounding geologic and hydrologic conditions were suitable, migration of waste to the accessible environment might be limited to acceptably low levels. On the other hand, thermomechanical disruption of the surrounding geology by the rock melt process might allow rapid transfer of contaminated ground water to surface aquifers, especially if promoted by thermal gradients from decay heat. While there is currently insufficient evidence to eliminate rock melt from further consideration on the basis of this standard of judgement, satisfactory performance appears highly uncertain. Furthermore a method for resolving this uncertainty does not appear to be available.

The host rock is the primary isolation mechanism for the shale-grout version of well injection. Assuming a suitably stable formation of adequate sorptive potential, preliminary calculations (Section 6.1.6) indicate that the likelihood of unacceptable quantities of radionuclides reaching accessible ground water is small. For dilute acid injection, assuming the site has suitable bounding formations, it also appears that there would be a low probability of unacceptable quantities of radioisotopes reaching accessible aquifers. However, prediction of acceptable long-term performance of well injection will require thorough characterization and understanding of the host formations and surrounding geology. It is highly uncertain at this time how this could be accomplished.

The transmutation concept may not require repositories providing 10,000-year isolation if all long-lived isotopes are eliminated. However, the 10,000-year isolation standard is not applicable to the transmutation process per se.

The space disposal concept appears to have most merit with respect to isolation. It has been calculated that a stable orbit would provide a minimum of 1 million years isolation.

In conclusion, it appears that all concepts with the exception of ice sheet, rock melt, and well injection have the potential of meeting the 10,000-year standard for acceptable waste isolation.

6.2.3.4 The Concept Should be Amenable to Development Within a Reasonable Period of Time Such That Implementation is Not Left to Future Generations

Necessary implementation time^(a) for the ice sheet concept is estimated to be 30 years or greater (Section 6.1.5) primarily because of the substantial uncertainties which remain to be resolved regarding ice sheet stability, structure, and dynamics and understanding of waste-ice interaction. A minimum time of 20 years is also projected for transmutation (Section 6.1.7); it is unlikely that this concept could be implemented prior to the turn of the century given the need to resolve theoretical uncertainties, and establish siting criteria; and the time required for pilot plant development, construction, and testing, and construction of commercial-scale facilities.

Development time has not been projected for the well injection concept. Although the engineering requirements for this concept do not appear difficult, requirements for improved site characterization techniques, performance assessment methods and monitoring technology appear to be formidable. However it may be possible to implement this concept within 20 years.

The remaining 20 years of this century would appear to be adequate for implementation of any of the remaining concepts, if it is assumed that very deep holes may be less than 10,000 m deep.

In summary, it appears that all concepts with the exception of ice sheet and transmutation qualify on this standard of judgement.

6.2.3.5 Implementation of a Concept Should Not Require Scientific Breakthroughs

Several concepts would require significant extension of existing technology to achieve satisfactory implementation; but none of the concepts appear to require scientific breakthroughs. Transmutation might be most efficiently accomplished in a fusion reactor, which would require a scientific breakthrough.

6.2.3.6 Capabilities for Assessing the Performance of a Concept Must Be Available Prior to Committing Major R&D Programs to Its Development

The need for substantial additional performance assessment capabilities appears to exist for all concepts. While the mined repository will require refinement of performance assessment capabilities, it is believed that this will be achieved in the near future. Manned inspection of the emplacement location is currently being proposed by the NRC. If this should be applied to all concepts, it would eliminate subseabed, very deep hole, ice sheet, well injection, space, and probably rock melt concepts.

All concepts, with the exception of transmutation, space, and subseabed require further development of remote sensing capability for assessment of the characteristics of the potential host media. In addition, the well injection and rock melt concepts would require

(a) All estimates of time assume that the concept discussed receives priority for funding.

development of methods for prediction and measurement of waste location and configuration. The lack of predictive methods for the ice sheet concept appears sufficiently intractable at this time to preclude consideration of this concept.

6.2.3.7 Implementation of a Concept Should Not Be Dependent Upon the Size of the Nuclear Industry

The rock melt, transmutation and space options appear to be potentially sensitive to the size of the nuclear industry. The reference rock melting concept would require sufficient waste product to operate at least one cavity (40,000 MTHM equivalent waste) and succeeding increments would be equally as large. The minimum size of a rock melt cavity has not been determined, however, and it is possible that smaller increments would be feasible. Transmutation would require operating reactors for the transmutation step and a sufficiently large industry to justify the investment in specialized support facilities. Space disposal, as well, would require a sizable investment in specialized hardware, needing a substantial nuclear industry to justify this investment. This, however, is an economic question and does not intrinsically disqualify space disposal from consideration.

6.2.3.8 Concepts Should Be Independent of Fuel Cycle Issues

Fuel cycles treated in this document include the once-through cycle and full uranium-plutonium recycle; however other cycles are possible. Although the uranium-only fuel cycle was discussed in the draft of this Statement, review comments indicate that this cycle is not considered reasonable by the industry or the scientific community and therefore this cycle is not considered further. Additional fuel cycle issues relate to timing of fuel cycle implementation and defense wastes.

Once-Through and Reprocessing Fuel Cycles

As summarized in Table 6.2.1, the mined repository and island mined repository concepts would be capable of accommodating all waste products of both the once-through and reprocessing fuel cycles. Various considerations suggest the use of mined repositories for bulky equipment and for the considerable volume of TRU wastes, hulls, and hardware generated by the reprocessing fuel cycle for disposal concepts that cannot accommodate these wastes.

The rock melt and well injection options could find application with either the once-through or the reprocessing fuel cycles. Fuel processing would be required for the once-through cycle.

The space disposal concept, as well, could find application to either fuel cycle, however, partitioning of the waste as well as processing of spent fuel would be required if the once-through fuel cycle were used.

Transmutation would find its most promising application with the reprocessing fuel cycle. Processing and partitioning of spent fuel and recycle in a reactor would be required and alternative disposal technology would be needed for disposal of other transmutation waste products, high-level liquid fission product waste and fuel hulls and hardware.

Timing

The timing of implementation of a waste management system could potentially affect the feasibility of the concepts because of declining decay heat generation rates or by the availability of facilities required to implement the concept. Substantial reduction of decay heat rates prior to emplacement of spent fuel or high-level waste could conceivably affect the operation of the rock melt and the ice sheet concepts; however reduction in decay heat rates over the time frames being considered for deferred fuel cycles do not appear to be great enough to materially affect operation of either of these concepts. Postponement of waste disposal operations beyond the period when light water power reactors were the dominant commercial type could impact the transmutation concept by requiring alternative transmutation devices. However, alternative devices, including fast breeder fission reactors and fusion devices, may be available and probably superior to light water reactors (Croff et al. 1980). Thus it is not felt that any concept can be dismissed on the basis of timing alone.

Summary of Fuel Cycle Issues

In summary, it appears that all of the concepts offer some potential benefit with any fuel cycle and that none should be dismissed because of sensitivity to fuel cycle issues (although the case for transmutation with a once-through fuel cycle appears to be quite marginal). Pursuit of the rock melt, well injection, transmutation or space disposal concepts with either fuel cycle would require concurrent development of one of the concepts capable of disposing of TRU waste, probably a mined repository.

6.2.3.9 Concepts Should Be Independent of Reactor Design Issues

None of the concepts appear to be especially sensitive to reactor design issues.

6.2.3.10 Implementation of a Concept Should Allow Ability to Correct or Mitigate Failure

This standard tends to favor those concepts in which wastes may be readily retrieved if observations of their actual behavior under full-scale implementation reveal previously unanticipated defects in the disposal system. Mined geologic disposal lends itself most readily to this requirement although obviously attempts at transmutation could easily be abandoned if large-scale operations failed to work.

Those concepts in which retrieval from a large-scale system would be difficult or impossible fail to meet this requirement. These concepts include space disposal, rock melt, well injection, and under certain circumstances, ice sheet disposal.

6.2.3.11 Maintenance or Surveillance Should Not Be Required for Extended Periods Following Termination of Active Repository Operations

The resolidification period of 1,000 years required of the rock melt concept would appear to require surveillance for a substantial period to verify long-term stability and satisfactory containment of the molten mass. This is seen as sufficiently contrary to this

standard of judgement as to prohibit preferred consideration of the rock melt option. The other concepts appear not to be affected by this consideration.

6.2.3.12 Summary

The performance of the nine proposed disposal concepts against the standards of judgement is summarized in Table 6.2.6. It should be emphasized that these conclusions are based largely on judgement of the authors, based in many cases on fragmentary or qualitative information. Of the nine proposed concepts, mined repository, very deep hole, island mined repository, subseabed, and space disposal have the potential for meeting all of the standards. A comparison of these five concepts is given in the next section.

6.2.4 Comparison of the Waste Disposal Concepts with Most Potential

This section compares the mined repository, island mined repository, very deep hole, subseabed and space disposal concepts on the basis of the assessment factors introduced in Section 6.2.2.

6.2.4.1 Radiological Effects

Operational Period

During the operational period, occupational exposure due to waste management would be dominated by that associated with waste processing. Transportation of TRU waste represents the greatest source of dose to the general public because of the large volume of material. Additional dose to both occupational workers and to the general public could result from accidents.

Occupational radiological effects attributable to processing operations would likely be quite similar for the mined repository, very deep hole, island mined repository, and subseabed options because the waste treatments are similar. Slightly greater occupational exposure could be expected with the very deep hole and subseabed options should it be decided to section bulky TRU-contaminated equipment for disposal by these options--an unlikely decision. Space disposal would require dissolution of spent fuel for both once-through and reprocessing fuel cycles, potentially resulting in greater radiological effects compared to the other options.

Transportation and handling requirements of spent fuel from power reactors to the waste treatment/packaging facilities would be approximately equivalent for each of the disposal concepts. The mined repository and very deep hole emplacement facilities could be colocated with the treatment/packaging facility so that no additional transportation is required. Alternately, the packaging facility could be located elsewhere. Subseabed would probably require two additional transport operations--transfer of waste packages to the embarkation port and subsequent ocean transport to the disposal site. Island repositories would require one additional movement, from the receiving port to the repository and would thus be equivalent to space disposal which would be characterized by a maximum of four major transport links for high-level waste. A smaller number of links could result from appropriate coloca-

TABLE 6.2.6. Performance of Proposed Waste Management Concepts on Ten Performance Standards

	<u>Radiological Standards</u>	<u>1,000-Year Containment</u>	<u>10,000-Year Isolation</u>	<u>Developmental Time</u>	<u>Scientific Breakthroughs</u>	<u>Predictive Capability</u>	<u>Industry Size</u>	<u>Fuel Cycles</u>	<u>Reactor Design</u>	<u>Ability to Correct or Mitigate Failure</u>	<u>Maintenance & Surveillance</u>
Mined Repository	X	X	X	X	X	X	X	X	X	X	X
Very Deep Hole	X	X	X	X	X	X	X	X	X	No	X
Rock Melt	X	No	No	X	X	X	X	X	X	No	No
Island Mined Repository	X	X	X	X	X	X	X	X	X	X	X
Subseabed	X	X	X	X	X	X	X	X	X	X	X
Ice Sheet	X	X	No	No	X	No	X	X	X	No	X
Well Injection	X	No	No	X	X	X	X	X	X	No	X
Transmutation	X	NA	NA	No	X	X	No	X	X	X	X
Space	X	X	X	X	X	X	X	X	X	No	X

X = The concept appears to have the potential to meet this standard based on available evidence.
 No = The concept does not appear to have the potential to meet this standard based on available evidence.
 NA = This standard is not applicable to this concept.

tion of facilities. The failure of a launch vehicle presents a potential single mode failure for space disposal and rapid rescue from incorrect earth orbit would likely be required to prevent public exposure.

Although, based on present evidence, any of the concepts could probably be conducted with radiation doses no greater than those currently permitted in fuel cycle facilities, substantial differences in cumulative radiation exposure might exist among the concepts. The above analysis suggests the following order of decreasing preference among concepts based on relative radiological effects during the operational period: mined repository; very deep hole; island mined repository; subseabed; space.

Post-Operational Period

Based on present evidence, any of the five concepts compared here has the potential to perform satisfactorily in the post-operational period (Section 6.2.3). However, probabilities of satisfactory performance differ and will be used as the basis of this comparison. Factors to be considered in evaluating the post-operational radiological integrity include failure of engineered containment to perform as expected, failure of natural barriers to perform as expected, compromise of repository integrity by catastrophic natural events exceeding design standards, and compromise of repository integrity by inadvertent human activity. From the standpoint of all four considerations, space disposal probably would provide the greatest certainty of satisfactory waste isolation in the post-emplacment period. In addition, the probability of satisfactory containment for several hundred years is seen as equally likely for the remaining concepts (see Section 6.2.3) although the performance of the package in the very deep hole is somewhat uncertain. Thus this discussion will focus on the prospects for longer-term isolation.

The effectiveness of natural barriers is seen to be potentially the greatest for the very deep hole concept because of the extreme depths involved. This assumes that depth alone will provide the single most effective barrier; however, uncertainties regarding the long-term integrity of the hole seal remain to be resolved. The mined repository concept relies on shaft seals as a barrier also but appears to offer greater probability of satisfactory long-term integrity due to the ability for human access during sealing operations. The possibility of disturbing the stability of the host sediment by emplaced waste might render the performance of the subseabed option less than that of mined geologic. The lack of understanding regarding behavior of island hydrologic systems under natural or waste-perturbed conditions raises significant questions as to the performance of the island mined repository in the long-term. For this reason the island mined repository concept is considered to be the least acceptable of the concepts on the basis of potential performance of natural barriers.

Of the four non-space concepts, very deep hole appears on the basis of its remote depth to offer superior protection from catastrophic natural events. Little distinction on this basis can be made between the subseabed, and mined repository concepts. Mined repositories on islands appear susceptible to catastrophic natural events associated with changes in future ocean levels.

As discussed in Section 6.2.1, efforts would be made to avoid siting repositories in areas having known or potential resource value, reducing the motivation for human intrusion. Fresh ground water can be a valuable resource in an island environment, however, and the presence of fresh water is intrinsic to the most potential island locations. Metal-bearing nodules are found--though they are scarce and of low grade--in the section of the ocean being considered for subseabed disposal. The resulting order of decreasing preference relative to prospects for inadvertent human intrusion would be space, very deep hole, mined repository, subseabed and island.

This overall analysis suggests the following order of decreasing preference relative to prospects for satisfactory radiological performance in the post-emplacment period: space; mined repository; very deep hole; subseabed; island mined repository.

6.2.4.2 Non-Radiological Environmental Effects

Health Effects

Implementation of any of the concepts would involve high-risk construction and operation activities including mining operations at sea and operations in space. Industrial accidents will undoubtedly occur; however, insufficient evidence currently exists to establish significant differences between options.

Injuries to the public could result from transportation accidents, and based on the number of transportation links inherent in each concept to which the public would be exposed (see Section 6.2.4.1), the order of decreasing preference would be the mined repository/very deep hole, island, and subseabed/space concepts. The mined repository and very deep hole concepts are essentially equivalent in this regard, as are the island and subseabed concepts.

Socioeconomic Effects

A comparative analysis of socioeconomic effects of generic disposal options is difficult because of the site specific nature of those effects. While one can assess factors such as size and number of facilities, the types of location and the size, timing and stability of the associated work force as discriminators among technology options, this is only half of the necessary information to assess impact. The other half consists of those factors associated with the area's ability to absorb the impacts. For example in times of high employment (no labor surplus) and high housing occupancy rates (no available housing) a project which requires high levels of manpower will create a serious (negative) impact. At a time when unemployment is high and housing is available, the same project would be of a positive impact.

Since these technologies involve different types of location and transportation steps, comparison against a "generic" location is not really possible. The addition of effects across several locations is not clearly a meaningful exercise since the impacts do not summate for any given community or person.

The mined repository and very deep hole disposal option would require only packaging plant and colocated repositories. Subseabed disposal would require a port facility in addition to packaging plants and the island concept would require, in addition, a receiving port and the island repository. The space disposal option would require processing, packaging, and launch facilities. An auxiliary waste disposal system for remotely handled and contact-handled TRU waste would likely be required for all concepts except mined geologic and island repositories.

In general, construction activities near small communities impact the socioeconomic structure of the community more than construction activities near large communities. Major facilities for the island geologic and subseabed disposal options would be located near the sea coast where the work force could typically be drawn from nearby communities. For the space disposal option, launch pad facilities exist and the required auxiliary facilities could be constructed at the launch site; however the waste treatment facility would also be required. The mined repository and very deep hole repositories would be located in areas of the continental United States, possibly in remote low population areas. In the case of space disposal especially there will likely be a substantial long-term increase in local employment due to the number of people required for support of launch activities. Subseabed has the same characteristics to a lesser degree, as does island disposal.

In conclusion, insufficient evidence (on a generic basis) is currently available to permit meaningful evaluation of alternative concepts on the basis of socioeconomic factors.

Aesthetic Effects

Aesthetic effects include noise, odors, and visual impacts. Analysis of aesthetic effects requires site-specific data because the effects are quite localized and dependent upon the design and siting of facilities. Because of this, characterization and comparison of aesthetic effects is not attempted in this Statement. Aesthetic effects would be an appropriate consideration in a statement considering proposed facility construction at a specific location. Items such as spoil piles from mined repositories and mud ponds from deep hole drilling could be unsightly, but the impacted area is not large.

Ecosystem Effects

Potential impacts of waste management facilities on ecosystems include effects on productivity, stability, and diversity. Evaluation of these effects at the generic level is difficult because of the sensitivity of these primary impacts to site and design characteristics which can only be addressed when considering specific installations. Consideration of such siting or design characteristics is beyond the scope of this generic statement. Thus to assess potential effects of the waste management options on ecosystems, it is necessary to look for effects inherent in the concepts under consideration.

Potential effects of the mined repository option include preemption of habitat during construction and operation of waste processing and repository facilities, potential releases of toxic waste processing chemicals to the environment and potential release of toxic spoil materials. Some preemption of habitat is unavoidable but with appropriate location and

design might well be limited to a few hundred acres of low productivity habitat. Release of toxic materials presents a potentially more severe problem. While it is predicted that release of chemicals from waste packaging facilities can be controlled to acceptable levels, control of spoils may prove difficult because of the open air storage required.

Very deep hole repositories would produce ecosystem effects similar to the mined repository option. Spoils, however, would be less bulky and presumably easier to control.

Island geologic, though technically similar to the mined repository concept, has a greater potential for ecosystem disruption because of the sensitive and unique characteristics of many island ecosystems. Assuming careful design and management of such a facility, however, the facility exclusion area might well protect or restore the integrity of the natural ecosystem as has happened to some extent at the sites such as the DOE site near Hanford, Washington. Leach of the spoil pile could significantly effect the quality of a small island ecosystem.

The potential ecological effects of the subseabed option are not known at this time. On-shore facilities are likely to be constructed near populated (and presumably ecologically disturbed) areas because of current efforts to protect what remains of natural coastline. A large area of seabed would be subject to penetrometer emplacement; however, the population and productivity of the affected region is likely to be low and relatively minor disturbance would be experienced.

Ecological effects of space disposal are likely to be modest (with the exception of those normally associated with space flight launches) in comparison to the other options. Assuming space disposal of all high-level waste, ancillary geological repository requirements would be very small compared to disposing of all waste in terrestrial repositories.

All concepts under consideration here offer the potential for satisfactory performance on the basis of non-radiological environmental effects; however, important differences in the absolute magnitude of these effects may exist. Some discrimination is possible on the basis of non-radiological health effects to the general public; however, the generic nature of the study and the early stage of development of most of the concepts provide tenuous discrimination among concepts on the basis of occupational (non-radiological) health effects and socioeconomic, aesthetic, and ecological effects. The order of decreasing preference based on available evidence regarding non-radiological environmental effects is: mined repository/very deep hole, subseabed/island, space.

6.2.4.3 Status of Development

Availability of Technology for Construction of System

There are considerable differences among the concepts with respect to the engineering development needed for implementation. Construction for the mined repository and island repository options would use well-tested existing technology, although for novel applications. The waste treatment technology required to support the mined repository concept is also well advanced, having been the focus of substantial development. Less is understood

relative to waste treatment and packaging requirements for an island mined repository, and considerable development activity might be required if the waste form and package concepts developed for mined repositories proved unsuitable for the island repository environment. The island concept would also require development of ocean transport and related transshipment facilities. Development of this equipment, however, is not viewed as particularly difficult, but largely an extension of existing technology.

The technology and methodology for siting geologic and subseabed repositories are developed to the point that they may be implemented. Space is unique in that the final location for disposition is not severely restricted by terrestrial concerns. Other options are poorly developed with respect to siting technology.

Implementation of the subseabed option, in addition to requiring development of the transshipment and ocean transport technology, would also require development of emplacement and emplacement monitoring technology, suitable waste form and packaging for the subseabed environment, and recovery technology for emplaced waste packages.

Space disposal would require development of a number of supportive technologies. Some (e.g., the space shuttle) are currently under development for other purposes and much of the remaining hardware represents extension of existing technology.

The very deep hole concept would require a significant extension of existing technology if the 10,000-m depth is required. Of the techniques available for making deep holes only rotary drilling has been used to develop wells to depths approaching those envisioned for very deep holes. Rotary drilling has been used for drilling to depths of about 9,000 m at bottom diameters of 6-1/2 inches--both shallower and of less diameter than postulated for the reference very deep hole concept. Deeper holes of larger diameter are thought possible but have not been demonstrated. It is quite possible that 10,000-meter holes will not be required by the concept. Other current limitations include casing to required depths and tensile strength of wire rope. In addition to technology related to making the very deep hole, development of a suitable waste form and packaging is required.

Availability of Technology for Adequate Performance Assessment

All of the alternative options appear to require further development of performance assessment and integrated safety and reliability analysis; however, the extent of such development is likely to be far greater with those concepts which have not received substantial attention, especially very deep hole, island mined repository, and space disposal. Fewer performance uncertainties appear to be associated with the subseabed concept; considerable research is underway on the deep ocean environment and the sediments are a homogeneous and probably fairly predictable environment. Fewest uncertainties appear to be associated with the mined repository concept largely because of the greater amount of research that has been accomplished on this concept.

The following order of decreasing preference is suggested relative to the current status of development of the concepts: mined repository; subseabed/island mined repository; space/very deep hole.

6.2.4.4 Conformance with Federal Law and International Agreements

The mined repository and very deep hole concepts could be developed without apparent conflict with Federal law or international agreements. A conflict may arise for the island disposal concept depending upon the island location. It would appear appropriate that the island be a possession of the U.S. Transport of large quantities of waste over international waters has the potential of generating adverse response.

Potential conflict of the subseabed disposal with existing law has been examined in some detail. The dumping of high-level radioactive waste is prohibited by the U.S. Marine Protection, Research and Sanctuaries Act of 1972, and therefore, would require Congressional action for implementation. The London convention of 1972, a multinational treaty on ocean disposal, addresses the dumping of contact-handled TRU and non-TRU waste. Dumping of high-level waste is prohibited; however the treaty's prohibition against dumping arguably does not extend to controlled emplacement of high-level waste into submarine geologic formations. EPA interprets the treaty as making subseabed disposal illegal.

Certain aspects of space disposal are addressed by existing treaties. The 1967 "Treaty on Principles Governing the Activities of States in the Exploration and Use of Outer Space Including the Moon and Other Celestial Bodies" prohibits waste disposal on the moon but does not rule out waste disposal in heliocentric orbit. Nations may object to the space disposal option because the waste would travel over their territory before being propelled from earth orbit. The 1972 "Convention on International Liability for Damage Caused by Space Objects" defines the responsibility for objects falling to earth on other countries. Consideration of such liability would be required.

In summary, the decreasing order of preference emerging from consideration of possible legal constraints on implementation of the five concepts is: mined repository/very deep hole; island; space; subseabed.

6.2.4.5 Independence from Future Development of the Nuclear Industry

Of the five concepts under comparison, space disposal appears to be most sensitive to the future development of the nuclear industry since it is considered that a substantial nuclear capacity will be required to justify the required investment (Section 6.2.3).

6.2.4.6 Cost of Development and Operation

Preliminary estimates of the cost of construction and operation for the mined repository, very deep hole and subseabed concepts appear in Section 6.1. These have been compiled and converted to unit costs (mills/kWh) in Table 6.2.7. Cost estimates for the island mined repository and the space disposal concept were insufficiently complete to permit reduction to a unit basis.

Of the available unit cost estimates, the very deep hole concept appears to be the most expensive with estimated costs of 3.0 mills per kilowatt-hour (1980 dollars), not a significant proportion of typical current new construction power costs (30 to 50 mills/kWh). Because these cost estimates are very preliminary and because even the most costly option

TABLE 6.2.7. Estimated costs of Various Disposal Options (1980 dollars)

	Research and Development Cost \$ millions	Pre-Disposal Cost, \$/kgHM		Repository Costs			Total Cost, mills/kWh ^(a,b,c)	
		Once-Through	Reprocessing	Construction, \$ millions	Operating, \$ millions/year	Decommissioning, \$ millions	Once-Through	Reprocessing
Mined Repository, 6,000 MTHM/yr	3,700	100	170	2,600	87	25	0.7	1.0
Very Deep Hole, 5,000 MTHM/yr	900	100	170	2,800	2,100	40	2.5	3.0 ^(d)
Island	NA ^(e)	150	190	NA	NA	NA	NA	NA
Subseabed, 5,000 MTHM/yr	NA	150	190	760	29	54	0.8	0.9
Space, per flight	NA	210	170	NA	46 ^(f)	4	NA	NA

(a) Does not include Research and Development costs.

(b) Construction and decommissioning costs amortized over 17 years @ 7%.

(c) Waste production rate is 38 MTHM/GW-year.

(d) Includes 0.2 mills per kWh for ancillary repository.

(e) NA = not available.

(f) \$ million per flight.

appears not to significantly impact the cost of electrical power, a cost comparison should not currently be assigned significant weight in this analysis. It should be noted that the cost estimates for all concepts essentially assume that no currently unanticipated questions will arise, which is probably an unlikely assumption.

6.2.4.7 Potential for Corrective or Mitigating Action

Prior to closure and sealing of access tunnels and shafts, mined repositories (including those utilized in the island disposal concept) would allow failure detection and permit retrieval of waste canisters. This system allows flexibility to future generations as to how long they might choose to leave the facilities open to inspection. Following closure, failure detection would be more difficult, although remote instrumentation could be installed for this purpose. Corrective action would be difficult (though possible) as the location of the waste would be known and access tunnels could be reopened. Detection of repository failure exemplified by unexpected concentrations of radionuclides could allow the mitigating actions of restriction of access to contaminated aquifers and other measures including evacuation of affected areas.

Complete corrective action capability for the island mined repository concept would require development of systems for locating and retrieving casks lost at sea in the case of the sinking of a transfer ship. A similar system would be required for the subseabed concept. Transponder devices would be fitted to the casks while enroute, and location and retrieval of an individual cask from the seafloor is considered feasible using existing equipment. However, loss of a ship with waste within the hull would severely complicate retrieval operations. Retrieval of emplaced canisters is considered to be feasible using existing overcoring technology, although retrieval of a large number of canisters would likely be very expensive.

Full corrective action capability for space disposal would require a deep-ocean payload retrieval system if system failure released radionuclides to the atmosphere. No corrective action would be possible. If failure of the space disposal system were to occur after achieving orbit, backup launch and orbit transfer vehicles, and some means for correction of improper orbit would be required. Each of these is under consideration as part of the space disposal concept, and if successfully developed (along with appropriate monitoring systems), would provide corrective action capability for most situations.

Corrective action with the very deep hole concept is thought possible only while the package is attached to the emplacement cable.

In summary, mined repositories appear to offer the greatest potential for corrective action. Subseabed appears also to provide reasonable potential for corrective action with the principal problem being retrieval of waste from a transport ship lost at sea. Island mined repositories present the combined difficulties and assets of the subseabed and mined repository concepts. Full corrective action potential appears to be achievable with space disposal for all situations except failure of the waste packaging system during launch or pre-orbital operations. Corrective action is thought not to be possible with the very deep

hole concept following package disengagement. The following order of decreasing preference relative to corrective action is thus suggested: mined repository; island mined repository; subseabed; space/very deep hole.

6.2.4.8 Long-Term Maintenance and Surveillance Requirements

None of the five concepts being considered here appear to require significant maintenance and surveillance activities during the post-operational period.

6.2.4.9 Resource Consumption

Preliminary estimates of selected critical resources for mined repository, very deep hole, subseabed and space disposal are provided in Table 6.2.8. Because of the very preliminary state of development of most concepts as reflected in the apparent inconsistencies among the estimates of Table 6.2.8, comparisons on the basis of these estimates would not be meaningful.

6.2.4.10 Equity of Risk

None of the concepts appear to have significant differences in this respect. Subseabed, ice sheet, island, and space disposal have the positive feature that no one must live in close proximity to the final disposal location. This creates the initial impression that the impact and risk are far less for those alternatives than for mined repositories. However a situation is established wherein the process of transportation of wastes is channeled through one location. A judgement of the equity of risk and impact resulting from the focus of transportation versus the focus of disposal is yet to be established.

6.2.5 Conclusions

Results of the comparisons on the assessment factors are depicted in Table 6.2.9 which shows the preference rankings of the five concepts (mined repository, very deep hole, subseabed, island repository, and space) on each of the assessment factors for which discrimination was found among the concepts. For each factor, the rankings of the five waste management concepts are plotted along a preference continuum, ranging from "most preferred" at the extreme left to "least preferred" at the extreme right. Concepts are clustered where no differences were observed.

6.2.5.1 Mined Repository

Examination of Table 6.2.9 supports selection of the mined repository concept as the waste disposal concept for preferred development. This concept is a "most preferred" concept on six of the seven comparisons of Table 6.2.9, ranking second on one consideration, "Radiological Effects During the Post-Operational Period." Here, the apparent length of isolation provided by space disposal results in the latter being preferred to mined repositories. An overall evaluation of the Radiological Effects attribute, however, might place

TABLE 6.2.8. Estimated Resource Commitments for Various Repositories

<u>Critical Resource</u>	<u>Mined Repository^(a,c)</u>	<u>Very Deep Hole^(b)</u>	<u>Subseabed^(b)</u>	<u>Space^(b)</u>
Aluminum, MT	220	13,000	13,000	83,000
Chromium, MT	--	14,000	14,000	5,000
Nickel, MT	--	7,500	7,500	2,000
Water, m	1,300,000	199,000,000	--	60,000,000
Natural Gas or Propane, m	11,500	10,000,000	10,000,000	10,000,000
Electricity, kWh	3,400,000,000	56,000,000,000	20,000,000,000	59,000,000,000
Petroleum ₃ Derived Fuel, m	5,300,000	6,000,000	5,100,000	1,500,000
Other Fuel, MT	--	--	--	4,800,000

(a) Highest consumption construction scenarios of Tables 5.4.2 and 5.4.3 added to operational values.

(b) Highest consumption scenario indicated of Section 6.1.

(c) Island mined repository has similar commitments.

TABLE 6.2.9. Summary of Preference Rankings

	<u>Most Preferred</u> → <u>Least Preferred</u>
Radiological Effects	
Operational Period	(MR) (VDH) (IMR) (SS) (S)
Post-Operational Period	(S) (MR) (VDH) (SS) (IMR)
Non-Radiological Environmental Effects	(MR, VDH) (SS, IMR) (S)
Status of Development	(MR) (SS, IMR) (S, VDH)
Conformance with Law	(MR, VDH) (IMR) (S) (SS)
Independence from Future Development of the Nuclear Industry	(MR, VDH, IMR, SS) (S)
Potential for Corrective or Mitigating Action	(MR) (IMR) (SS) (S, VDH)

KEY: MR = Mined Repository
 VDH = Very Deep Hole
 IMR = Island Mined Repository
 SS = Subseabed
 S = Space.

space disposal in an intermediate position below mined repositories because of the low ranking of space disposal on the basis of radiological effects during the operational period.

6.2.5.2 Subseabed

No clear preference emerges between the subseabed disposal concept and the island mined repository concept. However, because of significant uncertainties regarding the long-term radiological integrity provided by island geologic and hydrologic systems, subseabed appears to be superior to the island mined repository concept for continued development as an alternative to mined repository waste disposal. An additional advantage may be provided by subseabed's unique characteristics as a genuine conceptual alternative to mined repositories in comparison with island disposal, which is basically a variant (with additional uncertainties) of the mined repository concept. Uncertainties remain to be resolved concerning the long-term integrity of the emplacement media; development of transportation, emplacement and monitoring technology; resolution of potential international conflicts; and development of corrective action capabilities. Research will still be required, especially with the objective of resolving the waste isolation potential of the subseabed sediment. Should this capability be demonstrated conclusively, engineering development of the system could proceed.

6.2.5.3 Very Deep Hole

Although not possessing any clearly defined advantages over the mined repository concept on the basis of currently available evidence, the very deep hole concept ranks generally high on most of the assessment properties. Very deep hole offers potential for a high degree of geologic barrier performance in the post-operational period and some possibility of superior working conditions compared to mined repositories. A key issue is the value of manned in-situ examination of the actual placement location to understand the condition and environment into which the waste package is to be placed. Significant problems remain however, including the need for substantial development of drilling technology, improved understanding of the geologic environment at very deep hole depths, and analytical verification of the postoperational integrity of very deep hole repositories and performance of packages at the requisite temperature and pressure. Since deep hole technology is being developed for other reasons (e.g., for geopressured methane and for geothermal purposes) it is likely that increased information will be available regarding these uncertainties. An additional problem is the difficulty of providing adequate corrective action capability. Thus, the very deep hole concept, though having potentially superior characteristics to other alternatives, is also characterized by greater uncertainties. For these reasons, although continued development of the very deep hole concept as a long-term alternative to mined repositories is recommended, the priority of development is considered to be secondary to the subseabed concept. The considerations of potential problems with corrective action and the relatively unadvanced status of technology weigh heavily in this decision.

6.2.5.4 Space Disposal

The principal argument for space disposal is its promise for extraterrestrial disposal of selected radioisotopes; but substantial reservations exist concerning this concept. These include the potential radiological risk of the concept during the operational period, non-radiological health effects, potential conflicts with international law, and the difficulty of developing acceptable corrective action capabilities. Because of these conditions, priority development of space disposal as an alternative to mined repositories would appear to be unwise.

6.2.5.5 Island Disposal

The island disposal concept appears to present few advantages over the subseabed concept or the mined repository and is characterized by significant uncertainties regarding its potential for long-term isolation of waste. The principal potential advantage of island disposal is sociopolitical--it offers the possibility of a repository site remote from habitation and, thus, possibly of greater acceptability to the general public. Furthermore, the potential for international cooperation in establishing a repository at a "neutral" site might be presented by an island. Subseabed, however, offers the same advantages; thus the island concept would have merit only if the sociopolitical advantages were seen to be highly important, an appropriate island were available, and if the subseabed concept proved not to be technically acceptable. Because of these considerations, and because of great uncertainties regarding the waste isolation potential of island geology, development of this concept is not recommended.

REFERENCES FOR SECTION 6.2

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SYSTEM IMPACTS OF PROGRAM ALTERNATIVES

To assess and compare the impacts of implementing the three program alternatives addressed in this Statement (see Section 3.1), an analysis was made using a computer simulation of the complete waste management system functioning over the lifetime of a nuclear power system. This analysis considers the treatment and disposal of all post-fission high-level^(a) and TRU wastes (including decommissioning wastes), as well as gaseous and airborne wastes. All waste management functions are accounted for and all radioactive waste streams are tracked each year from origin through treatment, storage, transport and accumulation in a disposal repository. Both the example once-through cycle and the example reprocessing cycle described in Section 3.2 and Chapter 4 are analyzed.

7.1 BASIS FOR SYSTEM SIMULATION

To cover the range of potential impacts of program implementation, five different nuclear power growth cases are considered. In all cases, the nuclear capacity is assumed to consist of one-third BWRs and two-thirds PWRs. These cases were described in Section 3.2 and can be summarized as follows.

Case 1--Present Inventory. In this case, we consider only the amount of spent fuel estimated to be on hand, including in-core fuel, at the end of 1980; this is approximately 10,000 MTHM.

Case 2--Present Capacity. In this case, we consider the amount of spent fuel that would result from continued operation of the present 50 GWe of nuclear capacity over its expected normal life cycle to retirement after 40 years operation.

Case 3--250 GWe in Year 2000. In this case, nuclear power capacity grows to 250 GWe in the year 2000. All nuclear power plants operate for an expected normal life cycle of 40 years, and the last plant shuts down in 2040. It is intended to assess the waste management impacts over the complete life cycle of a nuclear generating system.

Case 4--250 GWe Steady State. This case follows the same growth curve, to 250 GWe in the year 2000, but then replaces retired capacity to maintain the 250 GWe capacity to the year 2040 when the case terminates.

Case 5--500 GWe in Year 2040. In this case, we assume the same 250 GWe growth by the year 2000 as in Case 3 but continue capacity additions to 500 GWe in the year 2040 when the case terminates.

The nuclear capacities for these cases are shown in Table 3.2.1 and Figure 3.2.3. The total electric energy generated in these five cases is shown in Table 7.1.1. Although power generation terminates in the year 2040 in all cases, waste management operations and decommissioning activities are continued until all wastes are emplaced in disposal facilities. In all cases, this is accomplished by the year 2075. The system simulation encompasses a

(a) High-level waste in this context includes spent fuel in the once-through cycle.

TABLE 7.1.1 Electric Energy Generated in Nuclear Power Growth Scenarios

<u>Case</u>	<u>GWe-Yr</u>
1	200
2	1,300
3	6,400
4	8,700
5	12,100

period from 1980 to 2075. In addition, the radioactivity inventory in the final repositories is followed over a million-year period. This provides an accurate representation of the radioactivity source term for hazard analysis. However, because of the very large uncertainties associated with long-term predictions of events that might result in some future radiological hazard, it is not considered useful to attempt predictions of radiological consequences for periods beyond about 10,000 years.

The objective of the system simulation was to identify the cumulative impacts of implementing the proposed program and to compare the range of impacts that would result from implementation of the proposed program, with those that could result from implementation of the alternative program or the no-action alternative. The three program alternatives were described in Section 3.1 and can be summarized as follows.

- Proposed Program. The research and development program for waste management will emphasize use of mined repositories in geologic formations capable of accepting radioactive wastes from either the once-through or reprocessing cycles. This program will be carried forward to identify specific locations for the construction of mined repositories.
- Alternative Program. The research and development program would emphasize the parallel development of several disposal technologies. This action implies an R&D program to bring the knowledge regarding two or three disposal concepts and their development status to an approximately equal level. At some later point, a preferred technology would be selected for construction of facilities for radiological waste disposal.
- No-Action Alternative. This alternative would eliminate or significantly reduce the Department of Energy's research and development programs for radioactive waste disposal. Under this alternative, existing spent fuel would be left indefinitely where it is currently stored and any additional spent fuel discharged from future operation of commercial nuclear power plants would likewise be stored indefinitely in water basin facilities either at the reactors or at independent sites.

The proposed program represents adoption of the interim planning strategy referred to in the President's statement of February 12, 1980, announcing a comprehensive radioactive waste management program for this nation. The President stated in part, "I am adopting an interim planning strategy focused on the use of mined geologic repositories capable of accepting both waste from reprocessing and unprocessed commercial spent fuel." Final

adoption of this strategy was to be subject to "a full environmental review under the National Environmental Policy Act" which this Statement satisfies. The President further stated, "We should be ready to select the site for the first full-scale repository by about 1985 and have it operational by the mid-1990s." Subsequent to the President's statement the Department of Energy published (on April 15, 1980) a Statement of Position on a proposed NRC rulemaking on storage and disposal of nuclear waste (DOE/NE-0007). DOE states in that document that implementation of the interim waste disposal strategy will result in the establishment of operating geologic repositories within the time range of 1997 to 2006. An exact date of operation, depending on a number of variables, will be determined by the outcome of existing programs. For example, if a site in bedded or domed salt is selected and licensing schedules recently forecast by the NRC staff are assumed, repository operation as early as 1997 could be achieved. However if a hard rock such as granite is selected, and if allowances are made for other uncertainties such as licensing proceeding delays and a requirement for more rigorous subsurface site characterization prior to site selection, initial repository operation could be as late as 2006. To cover additional contingencies such as an accelerated effort to open a repository or, at the other extreme, additional delays for reasons not yet foreseen, a range of repository startup dates from 1990 to 2010 is used here. The range of impacts is important in this simulation rather than the specific dates of repository startup.

Implementation of the alternative program would result in extending the time to operation of the first disposal system. This action implies a further period of research and development to bring the development status of the selected disposal alternatives to an approximately equal status with current knowledge regarding geologic disposal. At that time, a preferred technology would be selected and effort would be concentrated on developing this preferred technology with a program similar to the currently planned program for implementing geologic disposal. Thus a substantial time delay is inherent in this alternative.

In this system simulation, mined geologic repositories are used to represent the disposal method ultimately selected under the alternative program. This concept is the only one developed sufficiently to model impacts and costs reasonably well, and any alternative disposal concept that might be selected would only be selected if it did not have significantly greater impacts or costs. The primary effect of the alternative program implementation is the required interim storage for spent fuel or reprocessing wastes, the additional transportation to and from this storage and the impacts and costs for these operations. Benefits of the delay inherent in this alternative program include the processing and disposal of older and thus less radioactive and cooler wastes. Implementation of this alternative program is simulated by a range of repository startup dates from 2010 to 2030.

For the no-action alternative, indefinite storage of spent fuel in water basin facilities with no ultimate disposal has been assumed. It is also assumed that reprocessing would not be undertaken. Only the first three nuclear growth cases are considered because, without disposal, growth of nuclear power generation beyond the year 2000 does not appear credible.

The nuclear power growth cases and repository startup dates considered for the once-through cycle system simulation are shown in Table 7.1.2. A range of repository startup dates was used for the first three cases, that is, 1990 to 2010 representing the proposed program and 2010 to 2030 representing the alternative program. The 2010 startup provides both the last year of the range under the proposed program and the first year of the range under the alternative program. To simplify the analysis, only a single mid-range repository startup date, year 2000 representing the proposed program and 2020 representing the alternative program, was used for Cases 4 and 5. However, the same potential range as in the other cases should be inferred.

The nuclear power growth cases and reprocessing and repository startup dates considered for the reprocessing system simulation are shown in Table 7.1.3. Cases 1 and 2 were eliminated from consideration here because reprocessing was not considered to be credible under

TABLE 7.1.2. Repository Startup Dates Considered in the Once-Through-Cycle System Simulations

<u>Nuclear Power Growth Cases</u>	<u>Proposed Program</u>	<u>Alternative Program</u>	<u>No-Action Alternative</u>
1. Present Inventory Only	1990 to 2010 ^(a)	2010 ^(a) to 2030	None
2. Present Capacity Normal Life	1990 to 2010 ^(a)	2010 ^(a) to 2030	None
3. 250 GWe System by Year 2000 and Normal Life	1990 to 2010 ^(a)	2010 ^(a) to 2030	None
4. 250 GWe System by Year 2000 and Steady State	2000	2020	--
5. 500 GWe System by Year 2040	2000	2020	--

(a) These cases are identical under both the proposed and alternative programs.

TABLE 7.1.3. Reprocessing and Repository Startup Date Combinations Considered in the Reprocessing-Cycle System Simulations

<u>Nuclear Power Growth Cases</u>	<u>Proposed Program</u>		<u>Alternative Program</u>	
	<u>Reprocessing</u>	<u>Repository</u>	<u>Reprocessing</u>	<u>Repository</u>
3. 250 GWe System by Year 2000 and Normal Life	1990	1990	1990	2010 ^(a)
	1990	2010 ^(a)	2010	2010 ^(a)
	2010	2010 ^(a)	1990	2030
4. 250 GWe System by Year 2000 and Steady State			2010	2030
	2000	2000	2000	2020
5. 500 GWe System by Year 2040	2000	2000	2000	2020

(a) These cases are identical under both the proposed and alternative programs.

these low-growth conditions. The reprocessing cases are complicated by the added uncertainty for reprocessing startup. For Case 3, reprocessing startup in the time period 1990 to 2010 was considered in combination with repository startup dates of 1990 to 2010 for the proposed program and repository startup dates of 2010 to 2030 for the alternative program. As in the once-through cycle cases, the 2010 repository startup provides both the last year of the range under the proposed program and the first year of the range under the alternative program. To simplify the analysis, only mid-range dates were considered for Cases 4 and 5, that is, reprocessing startup in year 2000 in combination with repository startup in year 2000 representing the proposed program and in year 2020 representing the alternative program. However, the same potential range as in Case 3 should be inferred.

In selecting reprocessing startup dates, it was assumed that even if the current moratorium on reprocessing were lifted immediately, at least 10 years would be required to complete the construction, licensing, and startup of a reprocessing facility. Since a considerably longer time period could conceivably be required before reprocessing could be initiated, the 2010 startup date was selected to illustrate the effect of reprocessing after a longer period of delay. The important factor here is not the reprocessing dates themselves, but the effect that a range of reprocessing startup dates has on waste management impacts.

7.2 METHOD OF ANALYSIS FOR SYSTEM IMPACTS

The information flow in the computer simulation used for this analysis is presented in Figure 7.2.1. The first two modules of this computer model (i.e., ORIGEN and ENFORM) were adaptations of existing programs (Bell 1973, Heeb et al. 1979), while the last two modules were developed specifically for this simulation.

The computer code ORIGEN (Bell 1973) was used to define spent fuel composition. The ORIGEN code calculates the average composition of the spent fuel discharged from a nuclear reactor based on a set of input parameters that characterize the irradiation conditions. The set of input parameters (i.e., neutron cross sections and spectral indices) used had been calibrated to match results of empirically measured spent fuel compositions. Isotopic data were calculated for 175 nuclides, including all significant fission products, activation products and actinides.

Twenty-eight ORIGEN cases representing both PWR and BWR fuel irradiations were used to describe the spent fuel compositions for all of the fuel cycle alternatives. These cases (see DOE/ET-0028, Sec. 10.1) include separate cases for each enrichment zone of the initial cores, a first reload and equilibrium reload fuel batch and three recycle fuel batches for both uranium and plutonium recycle. In addition, the low exposure fuel batches remaining when a plant is shut down for decommissioning are described. Whether recycling is used or not, all plants start up and shut down without recycle fuel in the core. Recycle of both uranium and plutonium is limited to equilibrium fuel reloads, and the amount of either recycle fuel in any year is limited to 50% of the equilibrium reload fuel.

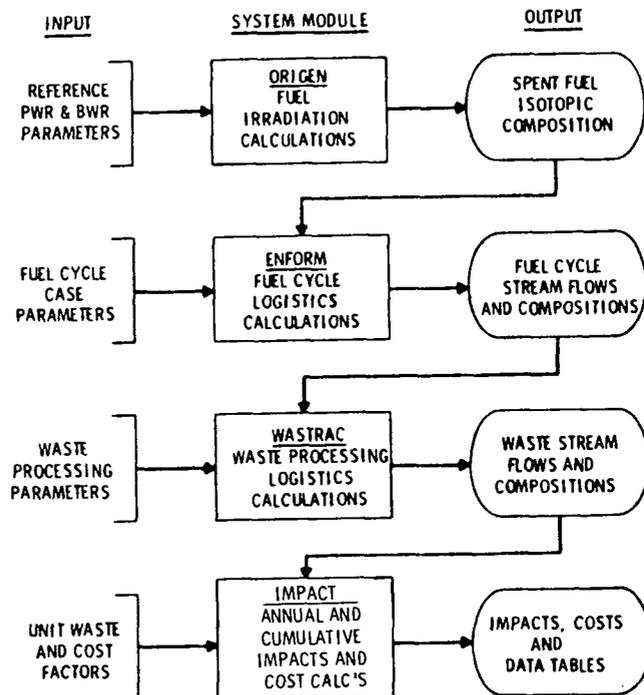


FIGURE 7.2.1. System Simulation Information Flow

By combining the ORIGEN to match the annual operating status of all plants in the system and the amount of uranium and plutonium available for recycle, the spent fuel composition with or without recycle in any year can be determined. This method of using a relatively small number of fuel irradiation (burnup) calculations to characterize a large number of spent fuel combinations provides an efficient and reasonably accurate representation of spent fuel compositions each year for the entire system.

The number of recycles for both uranium and plutonium was limited to three. The amount of third-recycle uranium and plutonium is small and the accumulation of ^{242}Pu in the third-recycle plutonium discharge reduces its value substantially. For these reasons and to simplify the calculation, the discharge from third-recycle fuel was discarded. In a real system whether or not the plutonium from the third recycle would be recycled would most likely be an economic decision. It could continue to be recycled and ultimately either be fissioned or transmuted to higher actinides and be discarded in the waste.

The computer code ENFORM (Heeb et al. 1979) was used to develop fuel cycle logistics and isotopic compositions of the fuel cycle streams. ENFORM was originally developed to evaluate environmental impacts of the entire nuclear fuel cycle. However, only its fuel cycle logistics capabilities were used here to provide fuel cycle source data for the WASTRAC module, which determined waste management logistics.

ENFORM input requirements include:

- a nuclear power growth projection
- a life-cycle operating schedule for the nuclear power plants
- recycle assumptions, i.e., once-through or recycle
- a fuel reprocessing schedule if recycle is selected
- inventory and timing assumptions for the entire fuel cycle
- spent fuel compositions as calculated by ORIGEN.

The output of the logistics calculation is a year-by-year mass flow and isotopic composition for each operation in the fuel cycle.

The computer code WASTRAC, developed for this analysis, models the storage, treatment, packaging, shipment and disposal operations for each waste stream. Figure 7.2.2 illustrates the waste management steps and the items calculated in a typical WASTRAC subsystem. Waste management steps can be added or deleted as required to model a specific subsystem. Each waste stream was tracked through a series of steps similar to that displayed in Figure 7.2.2.

WASTRAC computes waste volume and waste composition as a function of year, waste type and waste management step. The entire radionuclide content of the spent fuel is accounted for by allocating it either to a product stream, i.e., uranium or plutonium in a reprocessing case, or to one of the waste streams. Radionuclide inventories are corrected at each step for decay or buildup during the time interval since reactor discharge and/or reprocessing. Radionuclide inventories are also calculated for times up to one million years after placement in a final repository.

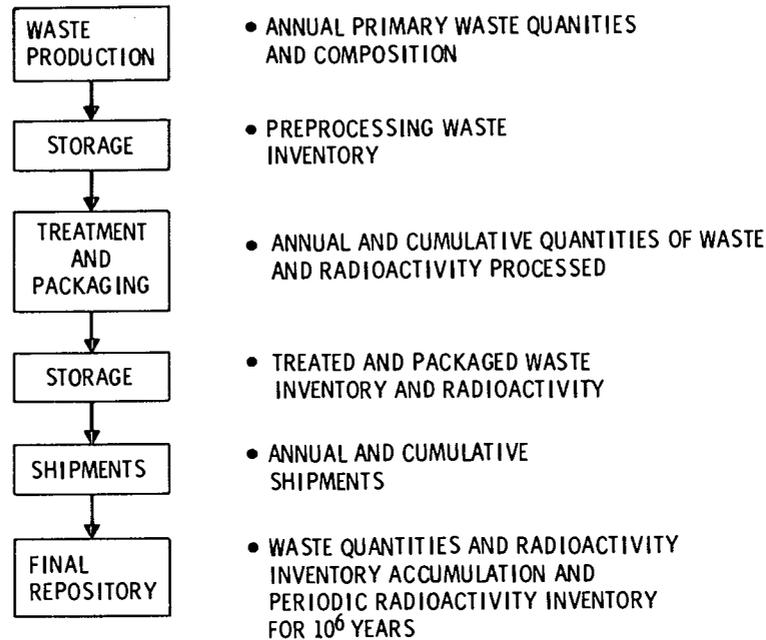


FIGURE 7.2.2. WASTRAC Calculations

The output of WASTRAC provides the waste volume and the quantity of each isotope in each waste stream at each step in the waste management system. Each treated waste stream is classified by container type and by the surface dose class for the treated TRU waste containers. Specifically waste streams are classified as high-level waste, remotely handled TRU (RH-TRU) waste (container surface-dose-rate equal or greater than 200 mrem/hr) or contact-handled TRU (CH-TRU) waste (container surface-dose-rate less than 200 mrem/hr).

The final step in the system simulation uses the time-dependent waste logistics data from WASTRAC to calculate the waste management impact and costs and to compile results in a series of tables. The computer code IMPACT was developed to perform these functions.

By utilizing release fractions for each isotope and each waste stream at each waste management step and dose factors per curie released, the isotopic releases and 70-year population radiation doses for each waste stream at each waste management step are calculated. Regional dose to whole body, bone, lungs, and thyroid and worldwide dose for release of ³H, ¹⁴C, and ⁸⁵Kr are calculated.

The IMPACT program organizes the results of the WASTRAC calculations, sums up annual and cumulative totals at specified intervals and prepares a series of tables to display the results. IMPACT also calculates both undiscounted and present-worth^(a) costs as well as levelized^(b) waste management costs per unit of power produced and per unit of fuel used.

(a) Present-worth discounting is a method of allowing for the time value of money. The present worth may be thought of as a present sum of money equivalent to a specified future payment or receipt or to a series of future payments or receipts. The present worth of a payment is obtained by multiplying the payment by $1/(1+i)^n$, where i equals the interest rate or discount rate and n is the number of years from the present to the time of the payment. The present worth of a series of payment is obtained by summing each payment's present worth.

(b) Levelizing refers to developing a single, constant unit charge, which recovers an expenditure associated with a facility or system including interest (see Section 3.2.8.2).

Four types of waste management costs are computed including treatment, interim waste storage, transportation, and repository costs. All costs are based on estimated unit costs as described in Sections 4.9 and 5.6. The cost of high-level waste treatment reflects an adjustment of high-level waste volume per container as limited by the thermal criteria at the geologic repository and the thermal energy of the waste at the time of emplacement.

Figure 7.2.3 schematically illustrates the relationship between the cash flow of the individual waste management system components and the discounting procedures. There are two similar but distinctly different applications of discounting techniques used in the development of the equivalent electric power and fuel cost of waste management. First, a present-worth leveling procedure is used to develop unit costs, i.e., cost per unit of spent fuel, for each waste management function. Second, a separate present-worth leveling procedure is used to convert waste management costs to equivalent electric power and fuel costs.

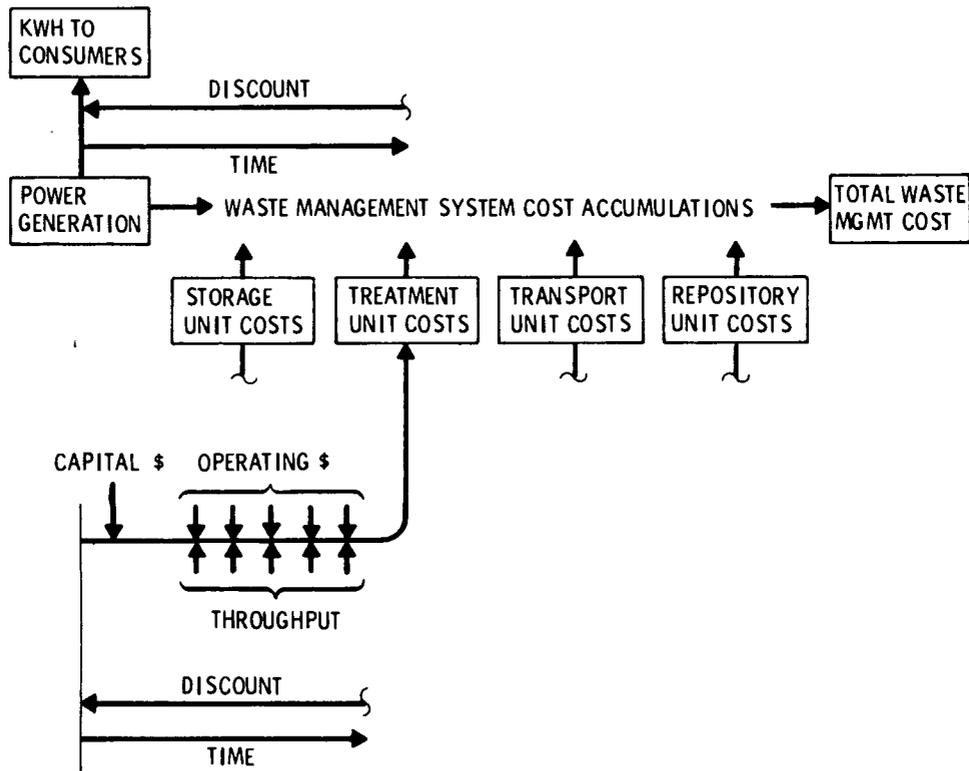


FIGURE 7.2.3. Time and Discounting Relationships of Waste Management Functions of Cost

The lower row of boxes in Figure 7.2.3 illustrates the functions that contribute to the total waste management system costs. The additional detail under the treatment unit-costs box indicates the flow of dollars and materials that are factored into the development of unit waste management costs. For any single waste management function all of the cash flows are present-worth discounted to a common starting point. The levelized unit cost for that function is then calculated by the relationship:

$$\text{Unit Cost} = \frac{(\text{Sum of present-worth costs})}{(\text{Sum of present-worth throughput})}$$

The unit cost developed by this procedure represents the single charge that can be assessed for the waste management function over the life of the facility that will recover all expenditures plus a return (the discount rate) on any unrecovered investment during the life of the facility. The sum of all the separate waste management system unit costs represents the total waste management system unit cost.

The accumulation of the waste management costs over a period of time following generation of power is also illustrated in Figure 7.2.3. It is assumed that all waste management costs, whether the services are provided by private industry or by the government, will be borne by the consumers of the electric energy generated by the nuclear power facility. Thus, the waste management costs will be reflected as an increase in cost of power.

The equivalent power costs of waste management can be obtained by discounting the costs of the individual waste management functions to the time of power generation, summing them all and dividing by the kilowatt hours of electric energy produced during the irradiation of the fuel. In other words, money is assumed to be collected from the rate payers to cover the cost of waste management at the time the electricity is generated. The amount collected is somewhat less, depending on the discount rate, than the costs of waste management will be when it is actually incurred. This allows the utility to earn a return on this money during this period so that a sufficient fund accumulates to pay for the waste management costs at the time they are incurred. At any interest rate (discount rate) greater than 0%, fewer dollars need be collected from the rate payers than will be required to pay later waste management costs at the time they are incurred. The higher the utility discount rate, the lower the waste management costs become.

7.3 SYSTEM LOGISTICS

To develop the system logistics requirements, some assumptions were made regarding the characteristics of a future nuclear industry and its associated waste management systems. These assumptions are not intended to be predictions of the future; rather, they are intended to provide a basis for estimating a potential range of requirements over a broad range of possible future developments. The results are valid primarily in terms of potential ranges of values. In general, the assumptions are intended to be conservative; that is, they err in a direction that tends to overstate rather than understate potential requirements and impacts.

The assumptions made in developing the logistics requirements for the once-through cycle were as follows.

1. Spent fuel is stored for a minimum of five years at the reactor basins after which it can be shipped to a repository if one is available.
2. The maximum storage capacity at the reactor basins averages 7 annual discharges. This is based on the assumption that reactor basin capacity will be expanded, on the average, to provide capacity for at least 3 full cores. Retaining full-core discharge capability and considering 3 annual discharges per core for a PWR and 4 annual discharges per core for a BWR results in an average capacity for approximately 7 annual discharges. This assumption also results in away-from-reactor storage requirements that approximate the maximum requirements shown in a recent study when currently licensed expansion plans of the electric utilities are assumed to be implemented and full-core reserve is maintained (DOE/NE-0002 1980).
3. After reactor storage basin capacity is filled, excess spent fuel is shipped to an away-from-reactor (AFR) independent spent-fuel storage facility.
4. When a repository opens, spent fuel is sent to the repository on a first-in, first-out basis; that is, the oldest fuel is always sent to the repository first.
5. Repository receiving capacity is expanded according to the following schedule for the first 10 years:

<u>Year</u>	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>	<u>6</u>	<u>7</u>	<u>8</u>	<u>9</u>	<u>10</u>
Receiving Capacity, MTHM	700	1,300	2,000	2,000	2,000	2,700	3,300	4,000	4,000	4,000

After 10 years 2,000 MTHM capacity increments can be added annually as needed to meet the demand. This capacity does not necessarily represent a single repository, but may represent several repositories that are opened up sequentially. However, single repositories with receiving capability of at least 6,000 MTHM per year are considered feasible.

6. The distance from a reactor to an AFR storage facility is 1,000 miles.
7. The distance from either a reactor or an AFR facility to a repository is 1,500 miles.

8. Spent-fuel from reactors is shipped 10% by truck and 90% by rail (45% by a combination of truck and rail using intermodal casks that can be transported by truck for short distances to a rail siding where they are transferred to a rail car and 45% by rail-only) while shipments from AFR facilities are 100% by rail.

The assumptions made in developing the logistics requirements for the reprocessing cycle were as follows.

1. A minimum storage period for spent fuel at the reactor basin is one year and at the reprocessing plant is one-half year.
2. The maximum storage capacity at the reactor averages 7 annual discharges.
3. Fuel that cannot be stored at the reactor basins is shipped to AFR storage facilities.
4. The reprocessing plant receives and processes spent fuel on a first-in, first-out basis; that is, the oldest fuel is processed first.
5. Reprocessing capacity is expanded in a pattern similar to the repository receiving capacity except that here each capacity increment is intended to represent a separate plant. Each plant has a 2,000 MTHM per year capacity and the second and third plants are restricted to startups at 5-year intervals. Each plant has a two-year restricted-throughput startup period, i.e., 700 MTHM in the first year, 1,300 MTHM in the second year and 2,000 MTHM/year thereafter. After 10 years, the interval between plant startups is restricted to a 3 year minimum.
6. Solidified high-level waste is stored for 5 years at the reprocessing plant before shipment. TRU wastes can be shipped as they are packaged.
7. If a repository is not available to receive the reprocessing plant wastes, storage is provided for high-level waste and TRU wastes at a separate independent site.
8. When the repository opens, it receives the wastes on the basis of the oldest waste first at the same rate they are produced. After 10 years, the receiving rate is accelerated as necessary to eliminate the storage backlog at the end of the 30th year.
9. When interim storage is required, all wastes flow through the storage facility until the backlog is eliminated. This assures that the oldest waste is sent to the repository first.
10. Shipping distances for spent fuel to the reprocessing plant or interim storage and from interim storage to reprocessing are 1,000 miles. Treated waste shipment distances from reprocessing or MOX fuel fabrication plants to interim storage are also 1,000 miles.
11. Shipping distances from the reprocessing or MOX fuel fabrication plants or from interim storage to a repository are 1,500 miles.

7.3.1 Repository Inventory Accumulations

The total amount of spent fuel to be disposed of or reprocessed for each of the five growth assumptions is shown in Table 7.3.1. The relative quantities of spent fuel here are approximately the amount that would result from the quantities of generated energy shown in Table 7.1.1. The proportional relationship is not exact, however, because only in Cases 2 and 3 do all reactor plants complete their full normal-life cycles.

TABLE 7.3.1. Total Spent Fuel Disposal or Reprocessing Requirements

<u>Case</u>	<u>Nuclear Power Growth Assumption</u>	<u>Spent Fuel Discharged, MTHM</u>
1	Present Inventory Only	10,000
2	Present Capacity and Normal Life	48,000
3	250 GWe System by Year 2000 and Normal Life	239,000
4	250 GWe System by Year 2000 and Steady State	316,000
5	500 GWe System by Year 2040	427,000

Only the once-through cycle is considered for the first two (low-growth) cases. The accumulation of spent fuel in the final repositories for these two cases is plotted in Figure 7.3.1 for each of the three repository startup dates. The region between the first two curves represents the range of inventory accumulations possible for the proposed program while the region between the second and third curve represents the range of inventory accumulations for the alternative program.

The repository inventory accumulation for Case 3 using the once-through cycle is shown in Figure 7.3.2. With the reprocessing cycle, however, the repository inventory accumulation is a function of both the reprocessing throughput and the repository startup and

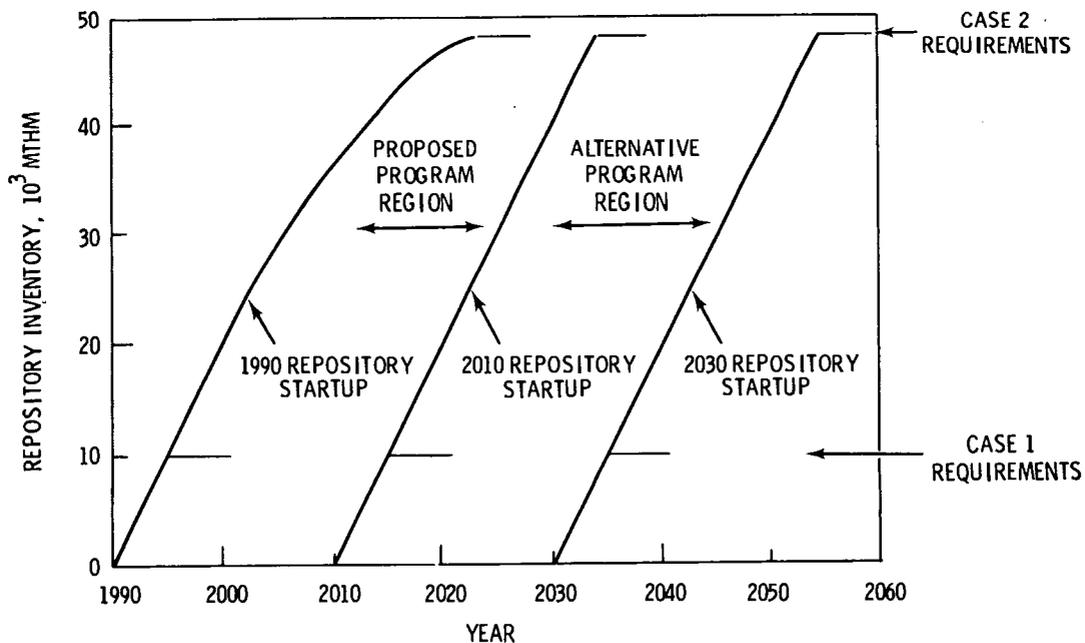


FIGURE 7.3.1. Repository Inventory Accumulations for Cases 1 and 2.

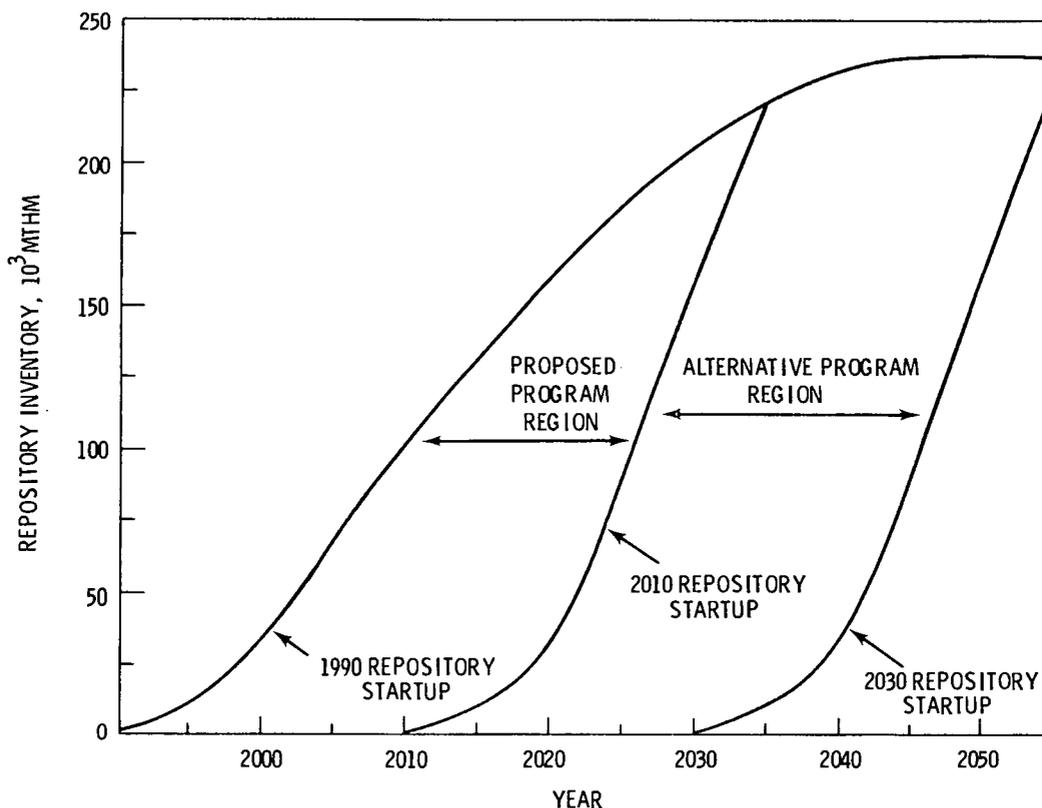


FIGURE 7.3.2. Repository Inventory Accumulation for the Once-Through Cycle in Case 3

receiving rates. The cumulative fuel reprocessed in Case 3 for the two reprocessing startup dates considered is shown in Figure 7.3.3. The repository accumulations of high-level wastes are plotted in Figure 7.3.4. Because of the five-year holdup of high-level waste at the reprocessing plant and because of the differences between the reprocessing rates and the repository receiving capacity, the high-level waste inventory accumulation in the 2010 repository is sensitive to the reprocessing date. For these reasons the region of inventory accumulation representing the proposed program and the region representing the alternative program overlap. The accumulation for the 2010 reprocessing startup and a 2010 repository startup forms the upper bound for the proposed program region while the accumulation for the 1990 reprocessing startup and a 2010 repository startup forms the lower bound for the alternative program region.

For Cases 4 and 5, only mid-range dates were used for reprocessing and repository startup dates. The repository inventory accumulation with the once-through cycle for Cases 4 and 5 are shown in Figure 7.3.5. The cumulative amounts of fuel reprocessed for Cases 4 and 5 are shown in Figure 7.3.6 while the repository accumulations of high-level waste are shown in Figure 7.3.7.

The total number of spent fuel canisters (see Section 4.3.1 for canister descriptions) sent to disposal with the once-through cycle is shown in Table 7.3.1a. Since the total quantity of spent fuel in a given case is the same for either the proposed or the

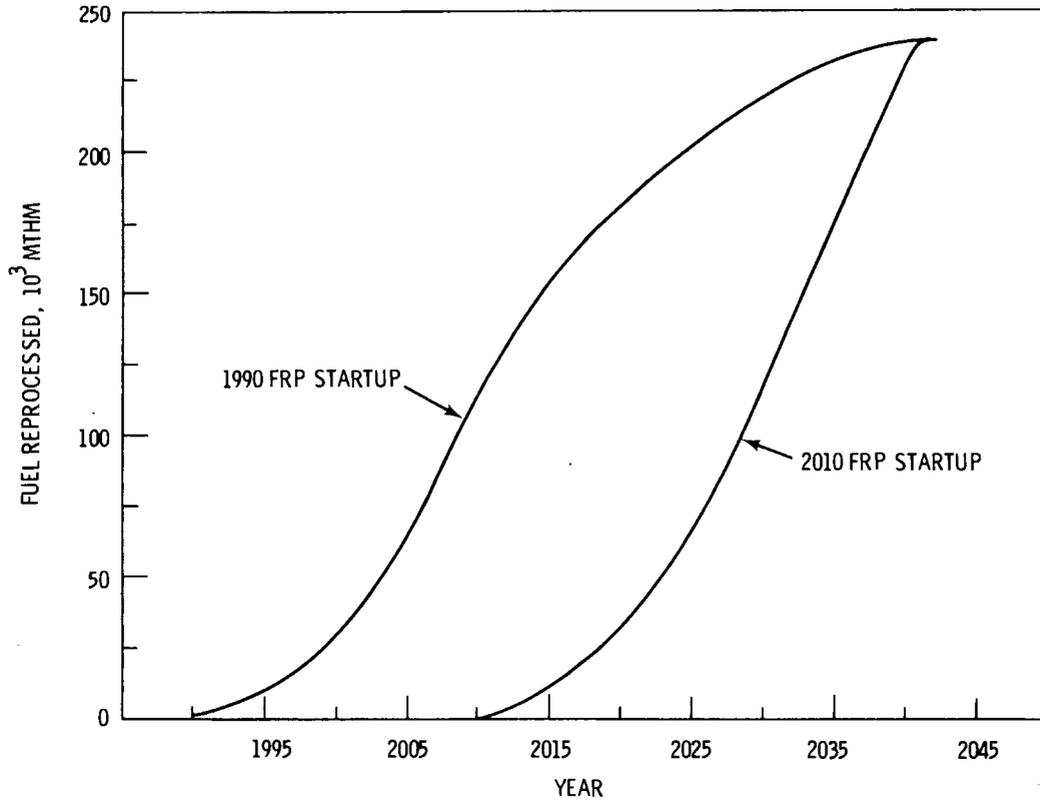


FIGURE 7.3.3. Cumulative Fuel Reprocessed for Case 3

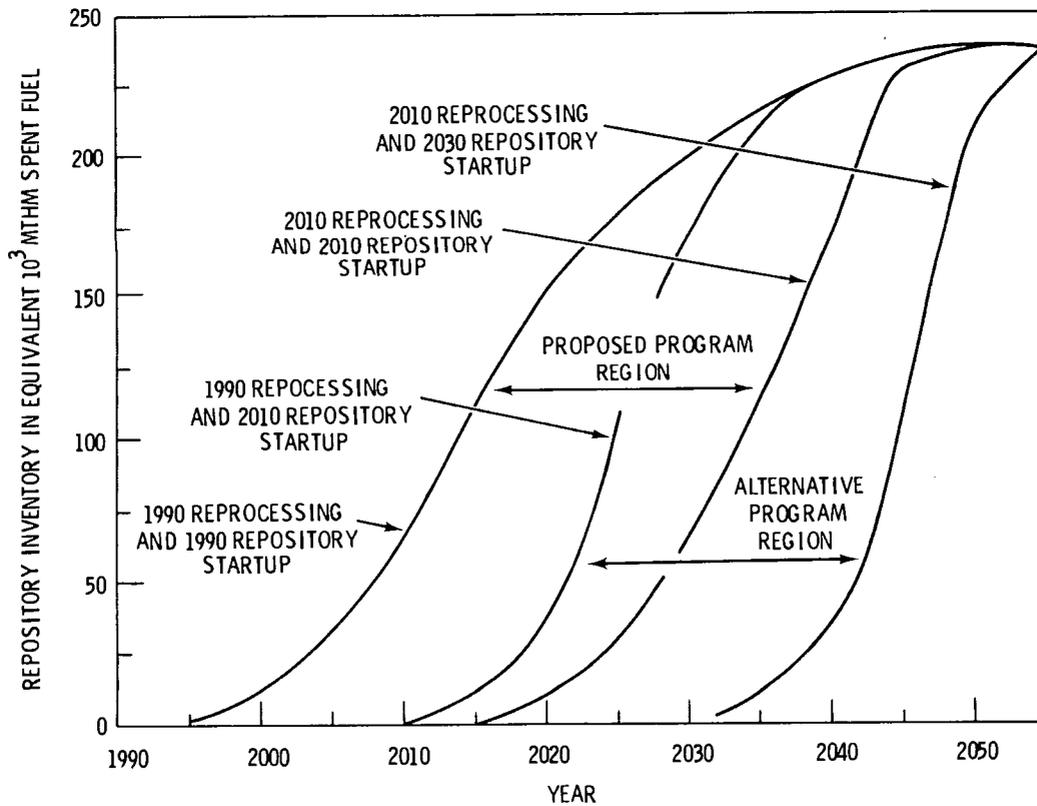


FIGURE 7.3.4. Repository High-Level Waste Inventory Accumulation for Case 3 with Reprocessing

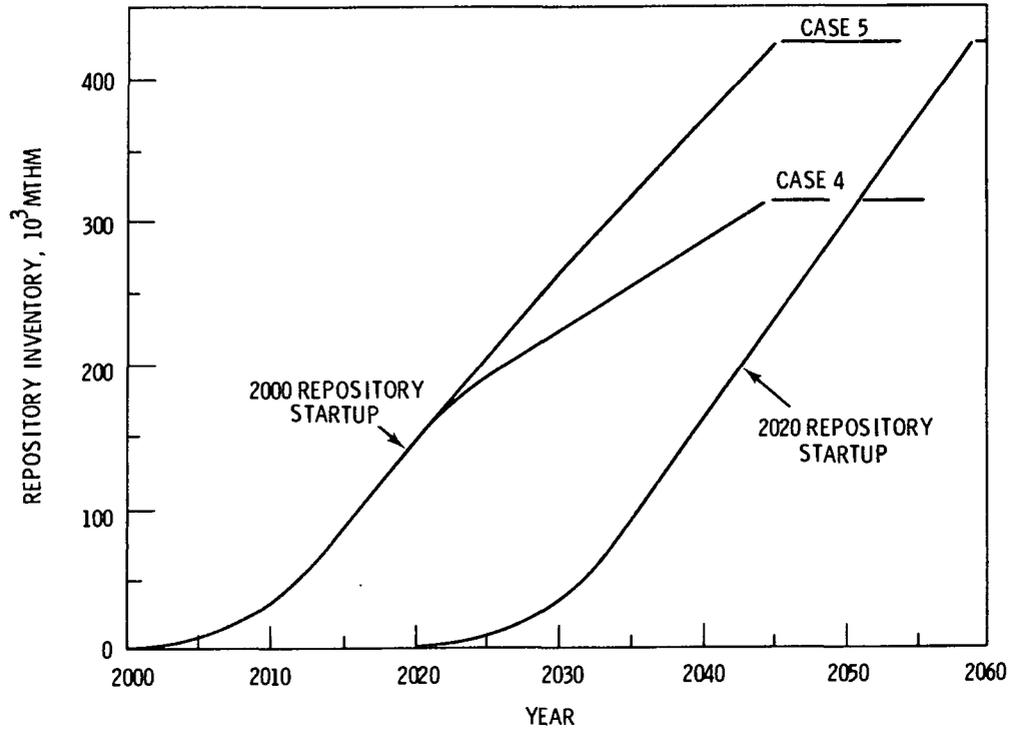


FIGURE 7.3.5. Repository Inventory Accumulation for The Once-Through Cycle in Cases 4 and 5

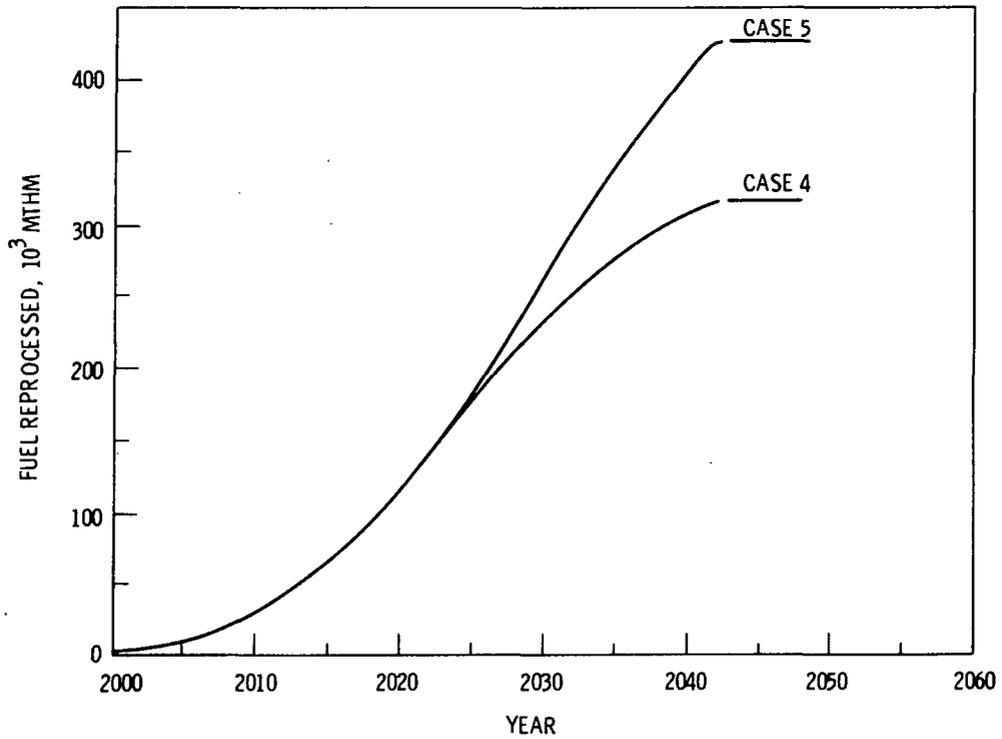


FIGURE 7.3.6. Cumulative Fuel Reprocessed for Cases 4 and 5

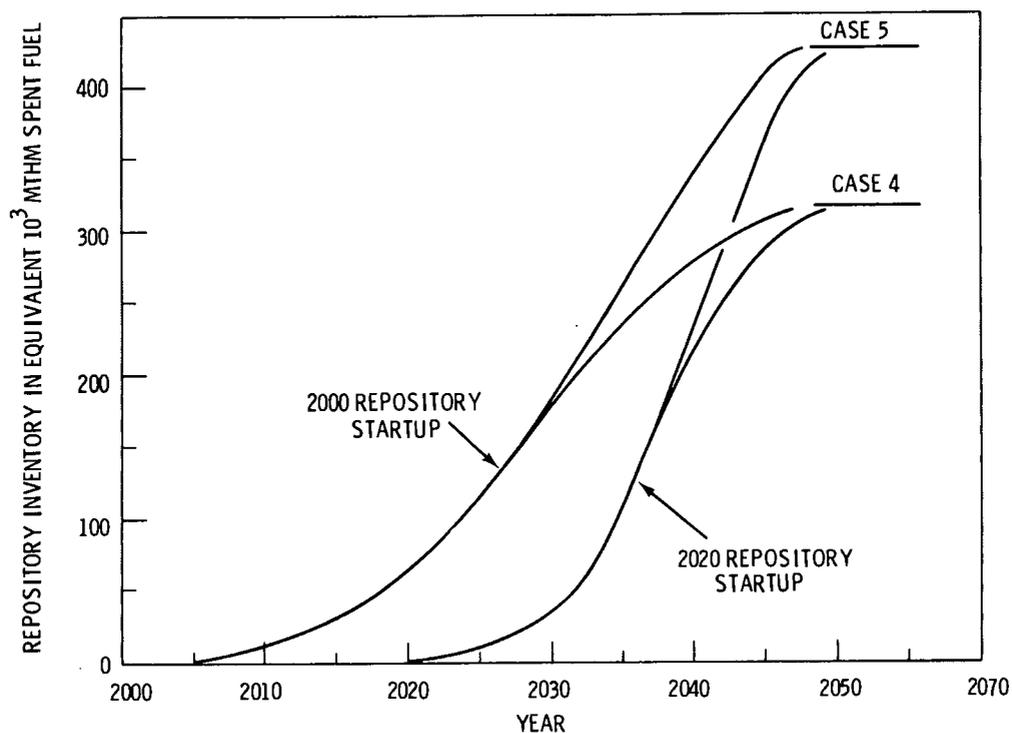


FIGURE 7.3.7. Repository High-Level Waste Inventory Accumulation for Cases 4 and 5 with Reprocessing

TABLE 7.3.1a. Number of Spent Fuel Canisters Sent to Disposal in the Once-Through Cycle

Case	Nuclear Power Growth Assumption	Thousands of Containers		
		Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)	No-Action Alternative
1	Present Inventory Only	35.6	35.6	0
2	Present Capacity and Normal Life	165	165	0
3	250 GWe System by Year 2000 and Normal Life	808	808	0
4	250 GWe system by Year 2000 and Steady State	1,070	1,070	NA(a)
5	500 GWe system by Year 2040	1,440	1,440	NA

(a) NA = not applicable.

alternative program and since we assumed that each fuel assembly would be encapsulated individually for this analysis, the number of canisters is the same for both major alternatives.

The total number of waste containers sent to disposal with the reprocessing cycle is shown in Table 7.3.1b (see Sections 4.3.2 and 4.3.3 for container descriptions). The range of numbers of high-level waste containers results from variations in the allowable heat

TABLE 7.3.1b. Number of Waste Containers Sent to Disposal in Reprocessing Cycle

Case	Nuclear Power Growth Assumption	Thousands of Containers	
		Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)
1	Present Inventory	NA(a)	NA
2	Present Capacity and Normal Life	NA	NA
3	250 GWe System by Year 2000 and Nor- mal Life		
	● HLW Canisters	80 to 430	80 to 180
	● RH-TRU Canisters	66	66
	● RH-TRU Drums	970	970
	● CH-TRU Drums	530 to 780	530 to 780
	● CH-TRU Boxes	9 to 11	9 to 11
4	250 GWe system by Year 2000 and Steady State		
	● HLW Canisters	140 to 350	114 to 270
	● RH-TRU Canisters	87	87
	● RH-TRU Drums	1,300	1,300
	● CH-TRU Drums	860	860
	● CH TRU Boxes	13	13
5	500 GWe System by Year 2040		
	● HLW Canisters	190 to 530	160 to 390
	● RH-TRU Canisters	117	117
	● RH-TRU Drums	1,740	1,740
	● CH-TRU Drums	1,200	1,200
	● CH-TRU Boxes	19	19

(a) NA = not applicable.

generation rate per canister for the four disposal media and variations in the age, and thus the heat generation rate, of the waste at the time of disposal. The contact-handled TRU waste quantities vary depending on the time reprocessing starts and the quantity of MOX fuel that is reprocessed. See Appendix Table A.1.22 for additional details.

7.3.2 Interim Storage Requirements

The interim storage requirements for spent fuel are controlled in the once-through cycle by the repository receiving capability, and in the reprocessing cycle by the reprocessing capacity. Spent fuel storage requirements in away-from-reactor (AFR) facilities, also referred to as independent spent-fuel storage facilities, are shown in Table 7.3.2 for the once-through cycle and in Table 7.3.3 for the reprocessing cycle. Requirements with or without reprocessing are about the same if repositories start up in the period of 1990 to 2010. However, whereas the storage requirements increase substantially for the once-through cycle with later repositories under the alternative program, the requirements are not changed in the reprocessing case since the storage requirement is controlled by the

TABLE 7.3.2. Comparison of Away-From-Reactor Spent Fuel Storage Requirements for the Program Alternative Using the Once-Through Cycle

Case	Nuclear Power Growth Assumption	Maximum Storage Requirements, MTHM		
		Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)	No Action Alternative
1	Present Inventory Only	0	0	0
2	Present Capacity and Normal Life	7,900 to 30,000	30,000 to 37,000	37,000
3	250 GWe System by Year 2000 and Normal Life	12,000 to 113,000	113,000 to 181,000	197,000
4	250 GWe System by Year 2000 and Steady State	60,000	176,000	NA ^(a)
5	500 GWe System by Year 2040	61,000	215,000	NA

(a) NA = not applicable.

TABLE 7.3.3. Comparison of Away-From-Reactor Spent Fuel Storage Requirements for the Program Alternative Using the Reprocessing Cycle^(a)

Case	Nuclear Power Growth Assumption	Maximum Storage Requirements, MTHM		
		Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)	No Action Alternative
1	Present Inventory Only	NA ^(b)	NA	NA
2	Present Capacity and Normal Life	NA	NA	NA
3	250 GWe System by Year 2000 and Normal Life	12,000 to 113,000	12,000 to 113,000	NA
4	250 GWe System by Year 2000 and Steady State	62,000	62,000	NA
5	500 GWe System by Year 2040	63,000	63,000	NA

(a) Assumed Reprocessing startup dates range from 1990 to 2010.

(b) NA = Not applicable.

range of reprocessing dates considered. The accumulation and decline of the storage requirements is illustrated for Case 3 in Figures 7.3.8 and 7.3.9 for the once-through cycle and reprocessing cycle, respectively. (See Appendix A.1 for annual requirements of other cases.)

Although in the reprocessing cycle the spent-fuel storage requirements are not increased by delay in repository availability, the storage requirements for the reprocessing

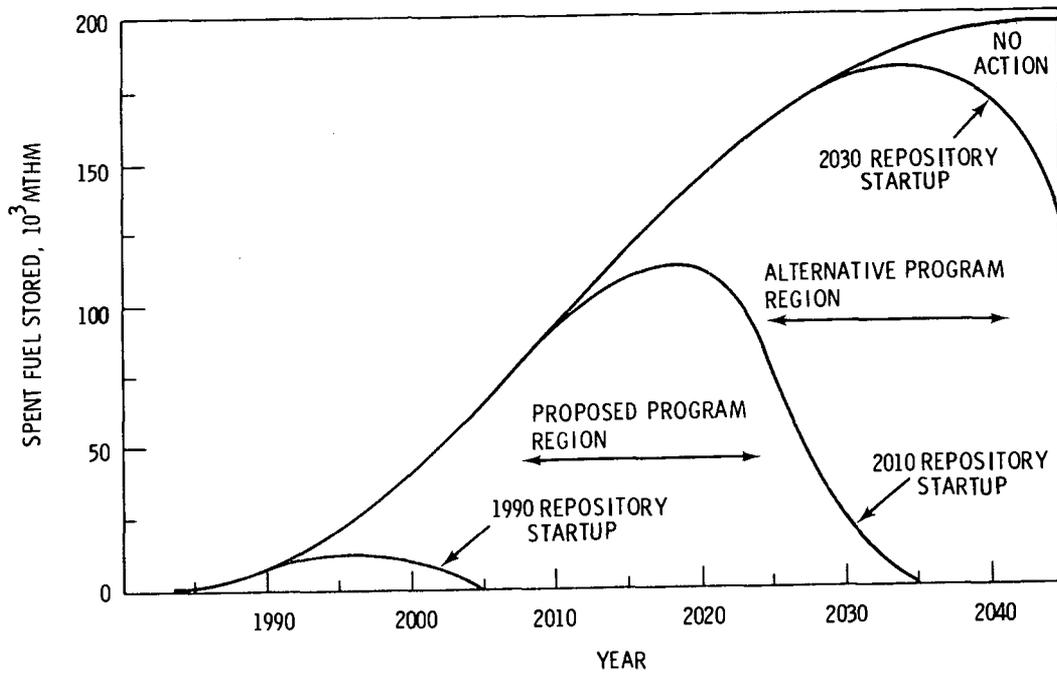


FIGURE 7.3.8. AFR Storage Requirements for Case 3 with the Once-Through Cycle

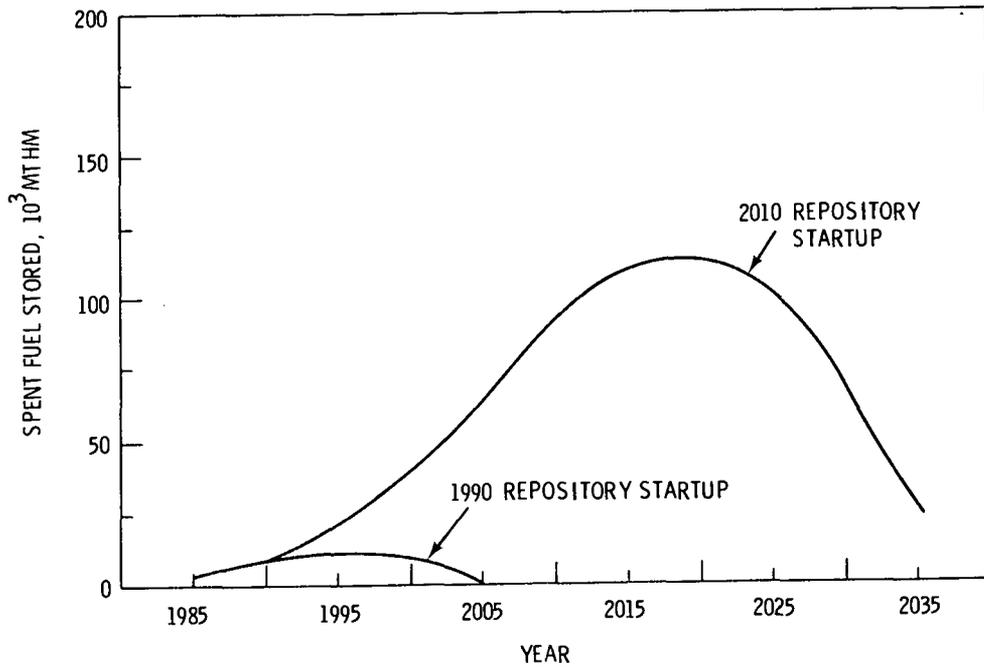


FIGURE 7.3.9. AFR Storage Requirements for Case 3 with Reprocessing

wastes do become substantial for delayed repository availability. This is shown in Table 7.3.4. The range of storage requirements for high-level waste canisters is affected not only by repository availability but also by the heat limitation on canisters for the different geologic media. For example, only about 1/3 as much high-level waste can be placed in a single canister for a repository in shale as can be placed in a canister for a repository in salt (see Section 5.3)..

TABLE 7.3.4. Interim Waste Storage Requirements for the Program Alternatives Using the Reprocessing Cycle^(a)

Case	Nuclear Power Growth Assumption	Maximum Number of Containers Stored		
		Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)	No Action Alternative
1	Present Inventory Only	NA ^(b)	NA	NA
2	Present Capacity and Normal Life	NA	NA	NA
3	250 GWe System by Year 2000 and Normal Life			
	● HLW Canisters	0 to 85,000 ^(c)	40,000 to 85,000 ^(c)	NA
	● RH-TRU Waste Canisters	0 to 41,000	41,000 to 60,000	NA
	● RH-TRU Waste Drums	0 to 604,000	604,000 to 894,000	NA
	● CH-TRU Waste Drums	0 to 397,000	337,000 to 577,000	NA
	● CH-TRU Waste Boxes	0 to 6,000	6,000 to 9,000	NA
4	250 GWe System by year 2000 and Steady State			
	● HLW Canisters	0	46,000 to 92,000 ^(c)	NA
	● RH-TRU Waste Canisters	0	54,000	NA
	● RH-TRU Waste Drums	0	798,000	NA
	● CH-TRU Waste Drums	0	460,000	NA
	● CH-TRU Waste Boxes	0	8,000	NA
5	500 GWe System by Year 2040			
	● HLW Canisters	0	52,000 to 114,000 ^(c)	NA
	● RH-TRU Waste Canisters	0	63,000	NA
	● RH-TRU Waste Drums	0	936,000	NA
	● CH-TRU Waste Drums	0	599,000	NA
	● CH-TRU Waste Boxes	0	10,000	NA

(a) Assumed reprocessing startup dates range from 1990 to 2010 (see Table 7.1.3).

(b) NA = not applicable.

(c) Range for HLW values for the four disposal media.

For Case 3 under the alternative program, the maximum storage requirements are not as large as one might at first expect considering the time delay to the year 2030 repository

startup. This is because of the declining schedule of fuel discharges (see Figure 3.2.3) and the accelerated repository receiving rate used to eliminate the storage backlog (see Figure 7.3.4). For Cases 4 and 5 under the proposed program, the repository starts the same year as reprocessing and there are no interim storage requirements. However, under the alternative program the storage requirements are substantial for these cases.

7.3.3 Transportation Requirements

Transportation requirements are identified here in terms of the number of shipments required. A shipment is defined as one truck cask or one rail or intermodal cask shipment in the case of spent fuel or one truck load or one rail car in the case of reprocessing wastes.

Transportation requirements for the once-through cycle are shown in Table 7.3.5. Truck shipments are the same under the proposed program or the alternative program. This is because it does not matter whether the fuel shipped from the reactor by truck goes to interim storage or the repository. It is only shipped once by truck as shipments from interim storage are assumed to be entirely by rail. Rail shipments can be higher under the alternative program because storage requirements are higher and any fuel shipped to interim storage must be shipped twice--once from the reactor to interim storage and once from interim storage to the repository. Fewer shipments are required under the no-action alternative because some of the fuel remains in the reactor basins and is not shipped at all. Additional details are shown in Appendix A, Table A.7.1.

Transportation requirements for the reprocessing cycle are shown in Table 7.3.6. Transportation requirements range somewhat higher under the alternative program than under the proposed program because more shipments are required to interim storage as a result of

TABLE 7.3.5. Comparison of Transportation Requirements for the Program Alternative Using the Once-Through Fuel Cycle

Case	Nuclear Power Growth Assumption	Transport Mode	Number of Spent Fuel Shipments		
			Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)	No Action Alternative
1	Present Inventory Only	Rail Truck	2,300 2,300	2,300 2,300	0 0
2	Present Capacity Normal Life	Rail Truck	13,300 to 18,000 11,000	18,000 to 19,000 11,000	8,400 8,600
3	250 GWe by Year 2000 and Steady State	Rail Truck	61,000 to 89,000 56,000	89,000 to 96,000 56,000	45,000 46,000
4	250 GWe System by Year 2000 and Steady State	Rail Truck	97,000 73,000	127,000 73,000	NA ^(a) NA
5	500 GWe by Year 2040	Rail Truck	126,000 99,000	170,000 99,000	NA NA

(a) NA = not applicable.

TABLE 7.3.6. Comparison of Total Transportation Requirements for the Program Alternative Using the Reprocessing Fuel Cycle^(a)

Case	Nuclear Power Growth Assumption	Transport Mode	Number of Shipments	
			Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)
1	Present Inventory Only	NA ^(b)	NA	NA
2	Present Capacity and Normal Life	NA	NA	NA
3	250 GWe System by Year 2000 and Normal Life	Rail Truck	90,000 to 119,000 182,000 to 314,000	117,000 to 147,000 182,000 to 317,000
4	250 GWe System by Year 2000 and Steady State	Rail Truck	136,000 250,000	176,000 412,000
5	500 GWe System by Year 2040	Rail Truck	179,000 343,000	233,000 566,000

^(a) Assumed reprocessing startup dates range from 1990 to 2010; (see Table 7.1.3)

^(b) NA = not applicable.

the potentially greater delay in repository availability. Requirements for truck shipments are much larger than in the once-through cycle because of the assumption that all TRU waste drums and boxes are shipped by truck. These wastes could be shipped by rail; in that case, only 1/2 to 1/3 as many shipments would be required. More details of the transportation requirements with the reprocessing cycle are shown in Appendix A, Table A.7.2.

7.3.4 Age of the Waste at Disposal

A potentially beneficial aspect of delayed repository availability under the alternative program is the aging of the waste, which reduces radioactivity and heat generation rates. The maximum and minimum ages at disposal for spent fuel from the once-through cycle and high-level waste from the reprocessing cycle are shown in Tables 7.3.7 and 7.3.8, respectively. To illustrate this aspect more fully, the ages of spent fuel and high-level waste for Case 3 are plotted as a function of time in Figures 7.3.10 and 7.3.11 for the once-through and the reprocessing cycles.

The lower thermal output for the aged waste would permit either more waste to be placed in individual canisters and a higher areal loading of the repositories, or could be used to provide a greater level of technical conservatism by allowing reduced temperatures for emplaced wastes. For this analysis, the quantity of high-level waste placed in individual canisters has been adjusted to take advantage of the lower thermal output of the aged waste, and the calculated repository requirements take into account the lower thermal output of the aged waste. The relationship between age of the waste and repository capacity is discussed in Section 5.3.3 and Appendix K.

TABLE 7.3.7. Maximum (and Minimum) Age of Spent Fuel Entering the Repository Using the Once-Through Cycle, Years

<u>Case</u>	<u>Nuclear Power Growth Assumption</u>	<u>Proposed Program (Geologic Disposal Starting 1990 - 2010)</u>	<u>Alternative Program (Disposal Starting 2010 - 2030)</u>
1	Present Inventory Only	18(14) to 38(34)	38(34) to 58(54)
2	Present Capacity and Normal Life	18(5) to 38(18)	38(18) to 58(38)
3	250 GWe System by Year 2000 and Normal Life	18(5) to 38(5)	38(5) to 58(19)
4	250 GWe System by Year 2000 and Steady State	28(5)	48(12)
5	500 GWe System by Year 2040	28(5)	48(20)

TABLE 7.3.8. Maximum (and minimum) Age of High-Level Waste Entering the Repository using the Reprocessing Cycle, (a) Years (b)

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)
1	Present Inventory Only	NA(c)	NA
2	Present Capacity and Normal Life	NA	NA
3	250 GWe System by Year 2000 and Normal Life	23(6.5) to 43(7)	38(6.5) to 58(13)
4	250 GWe System by Year 2000 and Steady State	33(6.5)	48(8)
5	500 GWe System by year 2040	33(6.5)	48(8)

- (a) Assumes reprocessing startup dates range from 1990 to 2010 (see Table 7.1.3).
 (b) Years from reactor discharge.
 (c) NA = not applicable.

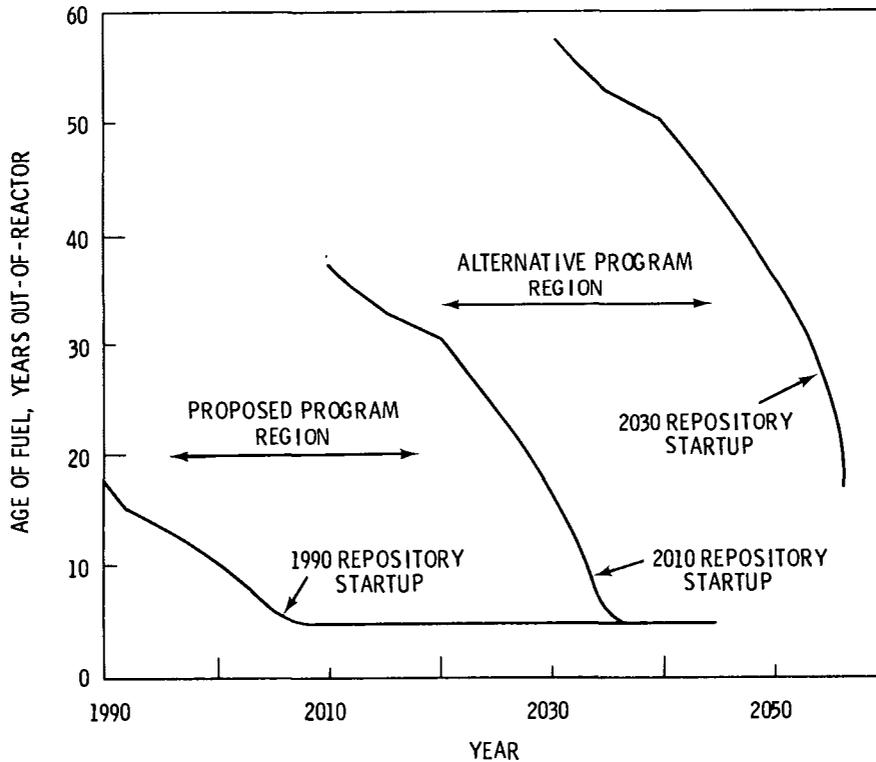


FIGURE 7.3.10. Age of Fuel Entering Repository for Case 3 with the Once-Through Cycle

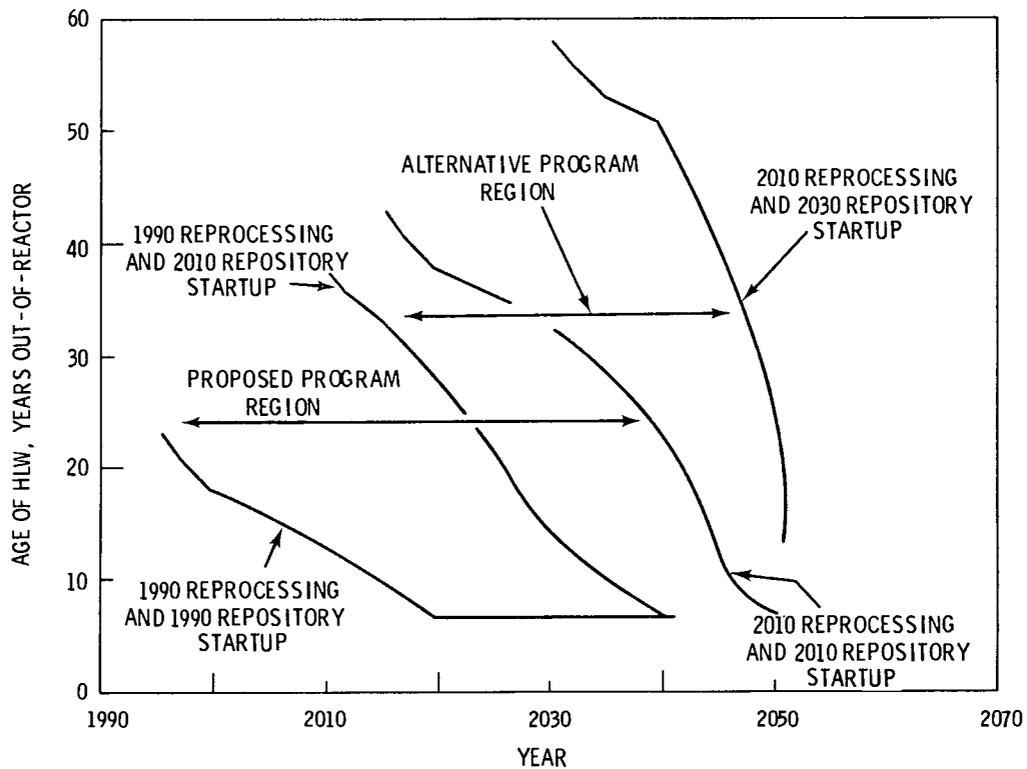


FIGURE 7.3.11. Age of High-Level Waste Entering Repository for Case 3 with the Reprocessing Cycle

7.3.5 Facility Requirements

To estimate resource requirements, it is first necessary to define the number of waste management facilities required in each case. In the once-through cycle, the only facilities required in addition to the repository and packaging facility are the independent fuel storage facilities for interim storage of the spent fuel. The number of these facilities required is proportional to the maximum spent fuel storage requirements shown in Table 7.3.2; a separate facility requirement table is not shown here. A 3,000 MTHM independent spent-fuel storage basin model was used in this Statement as a basis for resource requirement estimates. However, it is believed that facilities ranging up to 20,000 MTHM capacity might be used in cases where the interim storage requirements are very large. (Storage facilities up to 18,000 MTHM are considered in the U.S. Spent Fuel Policy Statement (DOE/EIS-0015 1980). The resource requirements and costs would decline somewhat as individual facility sizes increase because of scaling-effect efficiencies but radiation total releases would not be affected.

For the reprocessing cycle, the spent-fuel storage facility requirements would be proportional to the maximum storage requirement shown in Table 7.3.3. Other waste management facility requirements would be proportional to the number of fuel reprocessing plants and MOX fuel-fabrication plants utilized to process and recycle the spent fuel. Requirements for these facilities are shown in Table 7.3.9.

The number of equivalent 30-year-life plants utilized through the year 2040 was used to estimate resource requirements rather than number of plants started up. (Average utilization or capacity factor for a reprocessing plant was assumed to be 80% of on-stream design capacity and for a MOX fuel-fabrication plant a 65% factor was assumed.) It was assumed that the balance of the facilities started up would be utilized for continuing requirements outside the boundaries of the systems studied here. Both the number of startups and equivalent 30-year-life plants are shown in Table 7.3.9.

The number of repositories required is sensitive to the geologic medium. In the case of spent fuel, for example, the criteria utilized in this Statement indicate that the underground area required to store wastes in salt or shale is approximately twice that needed to store wastes in granite or basalt. For the reprocessing cycle wastes, salt compares favorably with granite and basalt, but shale requires on the order of twice the area required for the other three media examined. Taking into account the range of requirements for the four media considered here, Table 7.3.10 shows the range of 800-hectare (2,000-acre) repositories required for both the once-through and the reprocessing cycles. Further details can be found in Appendix Tables A.10.1 and A.10.2.

Although the range of requirements shown in Table 7.3.10 results largely from the range of geologic media considered, the range is also affected by the age of the waste. An older waste generates less heat and, as a consequence, permits somewhat more efficient use of repository space. The effect of waste age on repository capacity is discussed in Section 5.3.3.

Since significant improvements may yet be possible in both the once-through cycle repository concept and the reprocessing cycle repository concept, conclusions regarding relative repository requirements by fuel cycle should be considered as preliminary. The generally larger repository requirement for reprocessing wastes (salt is an exception) results from the additional placement area required for TRU wastes. (An illustration of the relative repository area requirements for each waste type can be found in DOE/ET-0028, Vol. 4, Tables 7.4.2 and 7.5.3.)

TABLE 7.3.9. Fuel Reprocessing and MOX Fuel Fabrication Plant Requirements

<u>Case</u>	<u>Nuclear Power Growth Assumption</u>	<u>2000 MTHM Startups</u>	<u>Fuel Reprocessing Plants Equivalent 30-yr-life plant Utilized</u>	<u>400 MTHM MOX Startups</u>	<u>Fuel Fabrication Plants Equivalent 30-yr-life-Plants Utilized</u>
1	Present Inventory Only	NA(a)	NA	NA	NA
2	Present Capacity and Normal Life	NA	NA	NA	NA
3	250 GWe System by Year 2000 and Normal Life	6	4	4 to 7	1.8 to 5.6
4	250 GWe System by year 2000 and Steady State	6	5.1	8	6.6
5	500 GWe System by year 2040	9	6.8	15	10.4

(a) NA = Not Applicable.

TABLE 7.3.10. Number of 800-hectare^(a) Repositories Required

Case	Nuclear Power Growth Assumption	Once-Through Cycle	Reprocessing Cycle
1	Present Inventory Only	0.03 to 0.1	NA
2	Present Capacity and Normal Life	0.2 to 0.7	NA
3	250 GWe System by Year 2000 and Normal Life	1 to 4	2 to 5
4	250 GWe System by Year 2000 and Steady State	2 to 5	3 to 6
5	500 GWe System by Year 2000	2 to 7	4 to 9

(a) 800 hectares = 2000 acres.

(b) NA = not applicable.

7.3.6 Equilibrium Requirements for Equilibrium Steady-State Systems

One of the purposes for Case 4 was to illustrate the level of continuing requirements in a steady-state nuclear system--in this case, 250 GWe. Table 7.3.11 shows these equilibrium requirements in terms of spent fuel disposal or reprocessing requirements, annual waste shipments and the number of years to fill an 800-hectare repository in the four geologic media. Requirements for other sizes of steady-state systems will be directly proportional to these requirements. For example, a 500 GWe steady-state system would have twice the requirements shown in Table 7.3.11. Data are provided on the number of years to fill repositories for waste ages of 5 and 50 years to lend perspective on the age variable. A significant improvement for the 50-year-old waste is indicated in all media for the reprocessing wastes and for spent fuel in granite or basalt, but relatively small improvements are shown for spent fuel in salt and shale.

TABLE 7.3.11. Equilibrium Requirements for Case 4 (250 GWe Steady State)

	Spent Fuel to Disposal or Reprocessing, MTHM	Annual Waste Shipments ^(a)		Time Required to Fill an 800-hectare Repository			
		Truck	Rail	5-yr-old Spent Fuel	50-yr-old Spent Fuel	5-yr-old HLW	50-yr-old HLW
<u>One-Through Cycle</u>							
Spent Fuel	6000	1400	1400				
Salt				10	11		
Granite				24	32		
Shale				12	15		
Basalt				24	32		
<u>Reprocessing Cycle</u>							
Spent Fuel	6000	1400	1400				
HLW and Other Wastes		3400	810				
Salt						11	21
Granite						11	23
Shale						6	11
Basalt						11	20

(a) A shipment is defined as one rail car or one truck load.

7.3.7 Plutonium Disposition

Examination of the disposition of plutonium helps to explain differences in the composition of the waste produced in the different nuclear growth cases and the effect that the reprocessing date has on the waste compositions (The reprocessing date effects the amount of recycle achieved within the time frame of this analysis.) Table 7.3.12 shows the plutonium disposition in both the once-through cycle and the reprocessing cycle. Disposition in the once-through cycle is straightforward--all of the plutonium goes to the repository with the spent fuel. With the reprocessing cycle, the situation is more complex. Much of the plutonium that is recycled is eliminated by fissioning. However, recycle of plutonium in mixed plutonium and uranium oxide fuel also produces more plutonium by conversion of ^{238}U . Thus, the total amount of plutonium generated in the reprocessing cycle is always larger than the total amount of plutonium in the once-through cycle spent fuel. Approximately 99% of the plutonium in the spent fuel is recovered by reprocessing and (excluding third-recycle discard) a little more than one percent of the plutonium ends up in the wastes; approximately 0.5% is in the high-level waste and the balance is dispersed in the TRU wastes. Plutonium recycle also produces more higher atomic number actinides (e.g., americium, neptunium and curium), which also end up in the waste.

At the end of the reactor operation period in each reprocessing case, there is some plutonium remaining in the fuel as well as plutonium in the reprocessing pipeline. This plutonium is shown in Table 7.3.12 as plutonium not recycled. It is assumed to be recovered by reprocessing but is not recycled in this system. We assume that other reactors that continue to operate outside of this system would, except for third-recycle plutonium, utilize this plutonium. Thus, except for the third-recycle portion, the plutonium not recycled is not considered for disposal in this Statement. Presumably, there will come a time when the industry will be shut down and the excess plutonium at that time will require disposal. However, before that time, steps could be taken to minimize the amount of plutonium left in the pipeline. With proper planning, the amount of plutonium requiring disposal could be reduced to the plutonium contained in the last batches of spent fuel. Since there would be no incentive for further reprocessing at that time, this spent fuel could be disposed of as spent fuel in the same manner as in the once-through cycle.

We assume here that the plutonium recovered from the third recycle is not recycled and that it is discarded in the high-level waste. Table 7.3.12 shows this to be a relatively small amount. In a real system, whether or not this plutonium is recycled will be primarily an economic determination. Recycle could be continued until all of the plutonium is either fissioned or transmuted to higher actinides, which are then discarded in the waste.

The two reprocessing dates used for Case 3 illustrate how sensitive the plutonium disposition is to reprocessing dates. Less than one-third as much plutonium is recycled when reprocessing starts in 2010 as when reprocessing starts in 1990. This is because of: 1) the large inventory of spent fuel accumulated when reprocessing starts, 2) a preference given to first-recycle plutonium relative to second- or third-recycle plutonium because of its higher fuel value, 3) the limitation on recycle MOX fuel to 50% of the equilibrium reload

TABLE 7.3.12. Plutonium Disposition Within the Timeframe of the Analysis

Case	Nuclear Power Growth Assumption	Once-Through Cycle Total Pu in Spent Fuel, MT	Reprocessing Cycle				
			Year Reprocessing Starts	Total Pu Generated, MT	Pu Recycled, MT	Pu Not Recycled, MT	Third Recycle Discard, MT
1	Present Inventory Only	36	NA(a)	NA	NA	NA	NA
2	Present Capacity and Normal Life	375	NA	NA	NA	NA	NA
3	250 GWe System by Year 2000 and Normal life	1,898	1990	3,429	2,308	1,121	152
			2010	2,160	644	1,516	0
4	250 GWe System by Year 2000 and Steady State	2,225	2000	3,779	2,146	1,633	0
5	500 GWe System by Year 2040	2,911	2000	5,147	3,584	1,559	0

(a) NA = not applicable.

fuel, 4) the long time for spent recycle fuel to work its way through the inventory to reprocessing, and 5) the year 2040 cutoff date for this analysis. No third-recycle fuel is irradiated in the Year 2010 reprocessing case. The same effect is noted in Case 4 and Case 5. We calculate that at equilibrium, 4 MT of third-recycle plutonium would be discharged for each 1,000 MT of equilibrium plus recycle reload fuel charged (equilibrium reload fuel accounts for approximately 80% of the total fuel). Thus, for Case 3 for example, where 239,000 MT of fuel are charged, the eventual implied commitment for third-recycle plutonium disposal is approximately 780 MT.

7.3.8 Radioactivity Inventory in Disposal Repositories

The total radioactivity and the total heat output from the entire inventory of all wastes sent to disposal from the entire system are summarized in Tables 7.3.13 through 7.3.16. These tables show the activity and heat output from year 2070 at periodic intervals for the next 1 million years for each of the nuclear growth cases. By the year 2070, all wastes have been placed in the repositories and much of the shorter life activities have decayed to low levels. Detailed tables showing the breakdown of radioactivity and heat output by individual nuclides are included in Appendix A.2 and A.3.

Table 7.3.13 shows the radioactivity inventory for all the fission and activation products. The radioactivity here is roughly proportional to the total energy produced in each case (see Table 7.1.1). The fission and activation product inventory for the reprocessing cases is closely similar to the fission and activation product inventory for the once-through cases.

Table 7.3.14 summarizes the total radioactivity inventory for all of the actinides and their daughter nuclides. The activity inventories in the once-through cases are roughly proportional to the energy generated in each case. This is also true for the reprocessing cases. However, the actinide inventories for comparable reprocessing and once-through cases are substantially different. The actinide activity initially is much higher with the once-through cycle wastes. This is because these wastes contain all of the plutonium present in the spent fuel. However, the recycle wastes contain a much higher level of the higher actinides--americum, curium, etc. Thus, the difference in total actinide activity inventories is not as large as one might expect based just on the plutonium content, and the differences become smaller in later years. Reprocessing Case 3 shows that the reprocessing date significantly effects the total actinide activity inventory in the wastes.

Table 7.3.15 shows total heat output for the fission and activation products and Table 7.3.16 shows heat output for the actinides and their daughter nuclides. These tables show that in all cases, the heat output is dominated by the actinides after the first 500 years.

Comparisons of the toxicity of radioactive wastes on the basis of hazard indices is discussed in Section 3.4. The relative toxicities of the once-through cycle and

TABLE 7.3.13. Total Radioactivity Inventory of All Fission and Activation Products in All Repositories(a)

Fuel Cycle	Case	Reprocessing Date	Curies								
			Year 2070	500 Years	1000 Years	5000 Years	10,000 Years	50,000 Years	100,000 Years	500,000 Years	1,000,000 Years
Once-Through	1	NA ^(b)	2.90×10^8	4.56×10^5	1.71×10^5	1.61×10^5	1.57×10^5	1.33×10^5	1.12×10^5	4.18×10^4	2.18×10^4
	2	NA	2.66×10^9	3.01×10^6	1.07×10^6	1.01×10^6	9.79×10^5	8.34×10^5	7.02×10^5	2.62×10^5	1.37×10^5
	3	NA	1.85×10^{10}	1.65×10^7	5.42×10^6	5.07×10^6	4.92×10^6	4.19×10^6	3.52×10^6	1.31×10^6	6.85×10^5
	4	NA	2.82×10^{10}	2.26×10^7	7.13×10^6	6.66×10^6	6.47×10^6	5.50×10^6	4.63×10^6	1.72×10^6	8.99×10^5
	5	NA	4.16×10^{10}	3.14×10^7	9.70×10^6	9.05×10^6	8.79×10^6	7.48×10^6	6.29×10^6	2.34×10^6	1.22×10^6
Reprocessing	3	1990	1.75×10^{10}	1.64×10^7	5.35×10^6	5.01×10^6	4.87×10^6	4.16×10^6	3.51×10^6	1.32×10^6	6.98×10^5
		2010	1.80×10^{10}	1.64×10^7	5.42×10^6	5.07×10^6	4.92×10^6	4.19×10^6	3.53×10^6	1.33×10^6	7.05×10^5
	4	2000	2.69×10^{10}	2.24×10^7	7.08×10^6	6.61×10^6	6.43×10^6	5.48×10^6	4.62×10^6	1.74×10^6	9.20×10^5
		2000	3.95×10^{10}	3.12×10^7	9.62×10^6	8.98×10^6	8.72×10^6	7.44×10^6	6.27×10^6	2.36×10^6	1.25×10^6

(a) Beyond 2070, time intervals are measured from 1980.

(b) NA = not applicable.

TABLE 7.3.14. Total Radioactivity Inventory of All Actinide and Daughter Nuclides in All Repositories(a)

Fuel Cycle	Case	Reprocessing Date	Curies								
			Year 2070	500 Years	1000 Years	5000 Years	10,000 Years	50,000 Years	100,000 Years	500,000 Years	1,000,000 Years
Once-Through	1	NA ^(b)	5.01×10^7	2.01×10^7	1.22×10^7	4.75×10^6	3.51×10^6	7.98×10^5	3.05×10^5	1.64×10^5	1.29×10^5
	2	NA	4.33×10^8	1.26×10^8	7.38×10^7	2.61×10^7	1.91×10^7	4.20×10^6	1.69×10^6	9.75×10^5	7.52×10^5
	3	NA	3.06×10^9	6.43×10^8	3.75×10^8	1.31×10^8	9.56×10^7	2.11×10^7	8.49×10^6	4.89×10^6	3.77×10^6
	4	NA	4.90×10^9	8.55×10^8	4.97×10^8	1.73×10^8	1.26×10^8	2.78×10^7	1.12×10^7	6.43×10^6	4.97×10^6
	5	NA	7.38×10^9	1.17×10^9	6.79×10^8	2.35×10^8	1.72×10^8	3.78×10^7	1.52×10^7	8.72×10^6	6.75×10^6
Reprocessing	3	1990	1.43×10^9	2.90×10^8	1.53×10^8	3.70×10^7	2.54×10^7	4.05×10^6	2.16×10^6	2.31×10^6	2.01×10^6
		2010	8.22×10^8	3.49×10^8	1.63×10^8	1.10×10^7	7.66×10^6	1.72×10^6	1.33×10^6	2.06×10^6	1.93×10^6
	4	2000	1.18×10^9	3.55×10^8	1.75×10^8	2.57×10^7	1.78×10^7	3.14×10^6	1.97×10^6	2.72×10^6	2.53×10^6
		2000	1.85×10^9	4.99×10^8	2.48×10^8	3.99×10^7	2.75×10^7	4.60×10^6	2.69×10^6	3.56×10^6	3.30×10^6

(a) Beyond 2070, time intervals are measured from 1980.

(b) NA = not applicable.

TABLE 7.3.15. Heat Output of Total Inventory of all Fission and Activation Products in All Repositories(a)

Fuel Cycle	Case	Reprocessing Date	Watts								
			Year 2070	500 Years	1000 Years	5000 Years	10,000 Years	50,000 Years	100,000 Years	500,000 Years	1,000,000 Years
Once-Through	1	NA ^(b)	8.85×10^5	6.79×10^2	2.95×10^2	2.83×10^2	2.76×10^2	2.29×10^2	1.84×10^2	4.13×10^1	1.09×10^1
	2	NA	8.18×10^6	4.53×10^3	1.87×10^3	1.79×10^3	1.75×10^3	1.45×10^3	1.16×10^3	2.59×10^2	6.84×10^1
	3	NA	5.72×10^7	2.50×10^4	9.41×10^3	8.99×10^3	8.77×10^3	7.28×10^3	5.82×10^3	1.30×10^3	3.43×10^2
	4	NA	8.73×10^7	3.43×10^4	1.24×10^4	1.18×10^4	1.15×10^4	9.56×10^3	7.65×10^3	1.71×10^3	4.51×10^2
	5	NA	1.29×10^8	4.80×10^4	1.68×10^4	1.61×10^4	1.57×10^4	1.30×10^4	1.04×10^4	2.32×10^3	6.12×10^2
Reprocessing	3	1990	5.38×10^7	2.60×10^4	9.98×10^3	9.53×10^3	9.29×10^3	7.67×10^3	6.10×10^3	1.31×10^3	3.48×10^2
		2010	5.55×10^7	2.51×10^4	9.53×10^3	9.10×10^4	8.87×10^3	7.35×10^3	5.87×10^3	1.30×10^3	3.47×10^2
	4	2000	8.31×10^7	3.53×10^4	1.29×10^4	1.23×10^4	1.20×10^4	9.91×10^3	7.90×10^3	1.72×10^3	4.56×10^2
	5	2000	1.22×10^8	4.94×10^4	1.76×10^4	1.68×10^4	1.64×10^4	1.35×10^4	1.08×10^4	2.34×10^3	6.20×10^2

(a) Beyond 2070, time intervals are measured from 1980.

(b) NA = not applicable.

TABLE 7.3.16. Heat Output of Total Inventory of All Actinide and Daughter Nuclides in All Repositories(a)

Fuel Cycle	Case	Reprocessing Date	Watts								
			Year 2070	500 Years	1000 Years	5000 Years	10,000 Years	50,000 Years	100,000 Years	500,000 Years	1,000,000 Years
Once-Through	1	NA ^(b)	1.26×10^6	6.53×10^5	3.91×10^5	1.46×10^5	1.08×10^5	2.38×10^4	8.21×10^3	3.69×10^3	2.90×10^3
	2	NA	8.53×10^6	4.10×10^6	2.37×10^6	7.98×10^5	5.83×10^5	1.25×10^5	4.51×10^4	2.22×10^4	1.71×10^4
	3	NA	4.40×10^7	2.10×10^7	1.20×10^7	4.00×10^6	2.92×10^6	6.25×10^5	2.26×10^5	1.12×10^5	8.59×10^4
	4	NA	5.84×10^7	2.79×10^7	1.60×10^7	5.29×10^6	3.87×10^6	8.25×10^5	2.98×10^5	1.47×10^5	1.13×10^5
	5	NA	7.99×10^7	3.81×10^7	2.18×10^7	7.20×10^6	5.26×10^6	1.12×10^6	4.04×10^5	1.99×10^5	1.54×10^5
Reprocessing	3	1990	3.15×10^7	9.06×10^6	4.60×10^6	8.98×10^5	6.26×10^5	1.11×10^5	5.36×10^4	5.46×10^4	4.79×10^4
		2010	2.63×10^7	1.14×10^7	5.28×10^6	2.45×10^5	1.73×10^5	4.29×10^4	3.05×10^4	4.84×10^4	4.58×10^4
	4	2000	3.66×10^7	1.13×10^7	5.43×10^6	5.75×10^5	4.06×10^5	8.15×10^4	4.65×10^4	6.41×10^4	6.01×10^4
	5	2000	5.60×10^7	1.58×10^7	7.65×10^6	8.94×10^5	6.30×10^5	1.21×10^5	6.43×10^4	8.37×10^4	7.83×10^4

(a) Beyond 2070, time intervals are measured from 1980.

(b) NA = not applicable.

reprocessing cycle wastes are compared in Table 7.3.17. The index employed here is the amount of water required to dilute one MTHM equivalent of the waste to drinking water standards (10 CFR 20) divided by the amount of water ($8.7 \times 10^7 \text{ m}^3$) required to dilute the original uranium ore to drinking water standards.^(a) An index of 1.0 means the toxicity hazard is equivalent to the original uranium ore. Detailed tables summing the dilution hazard-index for all of the significant fission and activation products and the actinides and their decay products are presented in Appendix A.4

The data in Table 7.3.17 show essentially equivalent relative hazard indices for all of the once-through cycle cases. Equivalence (index = 1) with uranium ore is reached after about 10,000 years.

Except at the beginning where they are closely similar, the reprocessing waste indices are somewhat lower than the once-through indices and reflect sensitivity to the amount of plutonium recycle achieved as identified by the reprocessing date. Equivalence with uranium ore is reached between 1000 and 2000 years after repository closure.

Nuclides that account for 90-plus percent of the hazard index are listed in Table 7.3.18 for several time periods. Only Case 3 is shown for the once-through cycle since all once-through cases are similar.

Initially, in both cycles, ^{90}Sr accounts for 95+% of the hazard index. At 1000 years the principal contributors in the once-through cycle are ^{241}Am , ^{240}Pu and ^{239}Pu and in the reprocessing cycle are ^{241}Am , ^{243}Am and ^{240}Pu . At 10,000 years the principal contributors in the once-through cycle are ^{239}Pu and ^{240}Pu , while in the reprocessing cycle they are ^{243}Am , ^{240}Pu and ^{239}Pu . For the 100,000- to 1,000,000-year period in the once-through cycle, ^{226}Ra and ^{210}Pb (both daughters of ^{238}U) are the principle hazards, while in the reprocessing cycle, the principle contributors include ^{229}Th , ^{129}I , and ^{237}Np in addition to ^{226}Ra .

It should be noted that although this index is one way to measure relative toxicity of the wastes it says nothing about the complex pathway for a release or the probability of actual release of these materials to the biosphere. This is discussed in Section 5.5.

(a) Based on 0.2% uranium ore and 3% ^{235}U fresh fuel.

TABLE 7.3.17. Hazard Index of Repository Waste Inventory Relative to 0.2% Uranium Ore. (a)

Fuel Cycle	Case	Reprocessing Date	Year 2070	500 Years	1000 Years	5000 Years	10,000 Years	50,000 Years	100,000 Years	500,000 Years	1,000,000 Years
Once-Through	1	NA ^(b)	2.29×10^2	5.39	3.14	1.11	8.51×10^{-1}	4.16×10^{-1}	4.30×10^{-1}	3.74×10^{-1}	2.33×10^{-1}
	2	NA	4.38×10^2	7.09	3.99	1.26	9.65×10^{-1}	4.89×10^{-1}	5.23×10^{-1}	4.38×10^{-1}	2.52×10^{-1}
	3	NA	6.14×10^2	7.29	4.08	1.27	9.73×10^{-1}	4.93×10^{-1}	5.27×10^{-1}	4.41×10^{-1}	2.53×10^{-1}
	4	NA	7.08×10^2	7.33	4.09	1.27	9.72×10^{-1}	4.88×10^{-1}	5.20×10^{-1}	4.37×10^{-1}	2.51×10^{-1}
	5	NA	7.73×10^2	7.43	4.14	1.28	9.77×10^{-1}	4.89×10^{-1}	5.20×10^{-1}	4.37×10^{-1}	2.52×10^{-1}
Reprocessing	3	1990	5.32×10^2	3.26	1.63	3.07×10^{-1}	2.24×10^{-1}	8.88×10^{-2}	7.91×10^{-2}	6.58×10^{-2}	3.15×10^{-2}
		2010	5.76×10^2	4.16	1.90	9.56×10^{-2}	7.21×10^{-2}	3.14×10^{-2}	2.78×10^{-2}	2.57×10^{-2}	2.08×10^{-2}
	4	2000	6.30×10^2	3.12	1.48	1.58×10^{-1}	1.16×10^{-1}	4.22×10^{-2}	3.66×10^{-2}	3.08×10^{-2}	2.21×10^{-2}
		2000	6.84×10^2	3.22	1.54	1.79×10^{-1}	1.32×10^{-1}	4.54×10^{-2}	3.88×10^{-2}	3.16×10^{-2}	2.21×10^{-2}

(a) Beyond 2070, time intervals are measured from 1980.

(b) NA = not applicable.

TABLE 7.3.18 Principal Contributors to the Hazard Index^(a)

Fuel Cycle	Case	Reprocessing Date	Year 2070		1000 Years		10,000 Years		100,000 Years		1,000,000 Years		
Once-Through	3	NA ^(b)	⁹⁰ Sr	95%	²⁴¹ Am	60%	²³⁹ Pu	52%	²²⁶ Ra	65%	²²⁶ Ra	68%	
			¹³⁷ Cs	2%	²⁴⁰ Pu	23%	²⁴⁰ Pu	38%	²¹⁰ Pb	22%	²¹⁰ Pb	23%	
					²³⁹ Pu	14%	²²⁶ Ra	4%	²³⁹ Pu	8%	²²⁹ Th	4%	
			<u>97%</u>		<u>97%</u>		<u>94%</u>		<u>95%</u>		<u>95%</u>		
Reprocessing	3	1990	⁹⁰ Sr	96%	²⁴¹ Am	75%	²⁴³ Am	34%	²²⁶ Ra	53%	²²⁶ Ra	30%	
			¹³⁷ Cs	2%	²⁴³ Am	11%	²⁴⁰ Pu	28%	²¹⁰ Pb	18%	²²⁹ Th	27%	
					²⁴⁰ Pu	10%	²³⁹ Pu	22%	¹²⁹ I	7%	¹²⁹ I	20%	
							¹²⁹ I	3%	²³⁹ Pu	6%	²¹⁰ Pb	10%	
							²²⁶ Ra	3%	²³⁷ Np	4%	²³⁷ Np	9%	
							¹²⁶ Sn	3%	²²⁹ Th	4%			
				<u>98%</u>		<u>96%</u>		<u>93%</u>		<u>92%</u>		<u>96%</u>	
		4	2000	⁹⁰ Sr	96%	²⁴¹ Am	94%	²⁴³ Am	40%	¹²⁹ I	23%	²²⁹ Th	41%
	¹³⁷ Cs			2%	²⁴³ Am	3%	²³⁹ Pu	19%	²²⁶ Ra	22%	¹²⁹ I	30%	
					²⁴⁰ Pu	12%	²³⁷ Np	13%	²³⁷ Np	13%	²³⁷ Np	13%	
							¹²⁹ I	9%	²²⁹ Th	10%	²²⁶ Ra	9%	
							¹²⁶ Sn	7%	¹²⁶ Sn	10%			
							²³⁷ Np	5%	²¹⁰ Pb	7%			
									²³⁹ Pu	7%			
				<u>98%</u>		<u>97%</u>		<u>92%</u>		<u>92%</u>		<u>93%</u>	
	5	2000	⁹⁰ Sr	96%	²⁴¹ Am	85%	²⁴³ Am	44%	²²⁶ Ra	33%	²²⁹ Th	36%	
¹³⁷ Cs			2%	²⁴³ Am	9%	²³⁹ Pu	19%	¹²⁹ I	17%	¹²⁹ I	29%		
				²⁴⁰ Pu	17%	²¹⁰ Pb	11%	²²⁶ Ra	14%	²²⁶ Ra	14%		
						¹²⁹ I	5%	²³⁹ Pu	9%	²³⁷ Np	12%		
						¹²⁶ Sn	4%	²³⁷ Np	9%				
						²³⁷ Np	3%	²²⁹ Th	8%				
								¹²⁶ Sn	8%				
			<u>98%</u>		<u>94%</u>		<u>92%</u>		<u>95%</u>		<u>91%</u>		

(a) Contribution of daughter nuclides is included.

(b) NA = not applicable.

7.4 SYSTEM RADIOLOGICAL IMPACTS

Both the regional and worldwide 70-year whole-body dose accumulations from normal operations for the proposed program, the alternative program, and the no-action alternative are compared for the once-through cycle in Table 7.4.1. Somewhat higher dose accumulations are indicated for the alternative program than for the proposed program. However, the differences are not large enough to be significant. The dose accumulation for the no-action alternative is somewhat less than for the other alternatives, but considering the time period involved, the differences are not significant. (There is a limit to how long spent fuel can be safely stored in water basins without further treatment. The assumption here is that this limit is not reached within the time frame of this analysis.) As would be expected, the dose increases with increasing size of the nuclear systems served.

TABLE 7.4.1. Comparison of 70-Year Whole-Body Dose Accumulations from Normal Operations for the Program Alternatives Using the Once-Through Cycle, man-rem

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal Starting 1990 - 2010)		Alternative Program (Disposal Starting 2010 - 2030)		No-Action	Alternative
		Regional	Worldwide	Regional	Worldwide	Regional	Worldwide
1	Present Inventory Only	36	48	36	48	0.2	4
2	Present Capacity Normal Life	200 to 250	290 to 370	250 to 260	370 to 380	90	160
3	250 GWe System by Year 2000 and Normal Life	940 to 1200	1400 to 1800	1200 to 1300	1800 to 1900	480	800
4	250 GWe System by Year 2000 and Steady State	1400	2100	1800	2600	NA(a)	NA
5	500 GWe system by Year 2040	1900	2800	2400	3400	NA	NA
	Dose Accumulation from Natural Radiation Sources	1×10^7	4.5×10^{10}	1×10^7	4.5×10^{10}	1×10^7	4.5×10^{10}

(a) NA = not applicable.

The regional and worldwide 70-year whole-body dose accumulations from normal operations for the proposed and alternative programs are compared for the case of reprocessing in Table 7.4.2. (The no-action alternative is not a consideration here because we assume that reprocessing would not be undertaken in that alternative.) The doses are much larger here than in the once-through cycle. However, considering the time period over which the dose is accumulated and comparing it to the dose to the regional and worldwide population that results from naturally occurring sources during the same period, 1×10^7 man-rem and 4.5×10^{10} man-rem, respectively, the dose is only a small fraction of the naturally occurring dose even in the highest nuclear growth case (Case 5); i.e., 0.5% of the regional dose

TABLE 7.4.2 Comparison of 70-Year Whole-Body Dose Accumulations from Normal Operations for the Program Alternatives Using the Reprocessing Cycle,^(a) man-rem

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal) Starting 1990 - 2010		Alternative Program (Disposal Starting 2010 - 2030)		No-Action Alternative	
		Regional	Worldwide	Regional	Worldwide	Regional	Worldwide
1	Present Inventory Only	NA ^(b)	NA	NA	NA	NA	NA
2	Present Capacity and Normal Life	NA	NA	NA	NA	NA	NA
3	250 GWe System by Year 2000 and Normal Life	13,000 to 33,000	580,000 to 970,000	13,000 to 33,000	580,000 to 970,000	NA	NA
4	250 GWe System by Year 200 and Steady State	33,000	1,000,000	33,000	1,000,000	NA	NA
5	500 GWe System by Year 2040	46,000	1,500,000	46,000	1,500,000	NA	NA
	Dose Accumulation from Natural Radiation Sources	1×10^7	4.5×10^{10}	1×10^7	4.5×10^{10}	1×10^7	4.5×10^{10}

(a) Assumed reprocessing startup dates range from 1990 to 2000.

(b) NA = not applicable.

and 0.003% of the worldwide dose. The doses from either the proposed program or the alternative program are identical. This is because the dose is accumulated primarily (about 95%) from the waste treatment operations and the same quantities of waste are treated in all cases--the only difference is that they occur at different times.

In this Statement, 100 to 800 health effects are postulated to occur in the exposed population per million man-rem. A health effect is either a fatal cancer or a genetic disorder. Based on this criterion, the program alternatives are compared on the basis of health effects in Table 7.4.3 for the once-through cycle and 7.4.4 for the reprocessing cycle. For the once-through cycle, even with the high nuclear growth assumption, the number of health effects range only from 0 to 2 on the regional basis and 0 to 3 on the worldwide basis. In the reprocessing case, the number of health effects are larger. For the high nuclear growth assumption, they range from 5 to 37 health effects on a regional basis and from 140 to 1100 on a worldwide basis. The health effects calculated to occur over the same period from naturally occurring radioactive sources range from 1000 to 8000 health effects to the regional population and 4×10^6 to 4×10^7 health effects to the worldwide population. Even though 140 to 1,100 may seem like a significant number of worldwide health effects, it is still only 0.003% of the calculated health effects to the worldwide population from naturally occurring sources of radiation over the same time period.

Neither the dose nor health effects comparison for normal operations provides a basis for favoring one of the program alternatives in either the once-through cycle or the reprocessing cycle. However, the potential impact of accidental releases might provide a basis

TABLE 7.4.3 Comparison of Normal Operations Health Effects for the Program Alternatives Using the Once-Through Cycle (number of deaths and/or genetic defects)

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal) Starting 1990 - 2010)		Alternative Program (Disposal Starting 2010 - 2030)		No Action Alternative	
		Regional	Worldwide	Regional	Worldwide	Regional	Worldwide
1	Present Inventory Only	0	0	0	0	0	0
2	Present Capacity and Normal Life	0	0	0	0	0	0
3	250 GWe System by Year 2000 and Normal Life	0 to 1	0 to 2	0 to 1	0 to 2	0	0 to 1
4	250 GWe System by Year 2000 and Steady State	0 to 1	0 to 2	0 to 1	0 to 2	NA(a)	NA
5	500 GWe System by Year 2040	0 to 2	0 to 2	0 to 2	0 to 3	NA	NA

(a) NA = not applicable.

TABLE 7.4.4 Comparison of Normal Operations Health Effects for the Program Alternatives Using the Reprocessing Cycle (number of deaths and/or genetic defects)

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal) Starting 1990 - 2010)		Alternative Program (Disposal Starting 2010 - 2030)		No-Action Alternative	
		Regional	Worldwide	Regional	Worldwide	Regional	Worldwide
1	Present Inventory Only	NA(a)	NA	NA	NA	NA	NA
2	Present Capacity and Normal Life	NA	NA	NA	NA	NA	NA
3	250 GWe System by Year 2000 and Normal Life	1 to 26	6 to 750	1 to 26	6 to 750	NA	NA
4	500 GWe System by Year 2040	3 to 27	100 to 800	3 to 27	100 to 800	NA	NA
5	500 GWe System by Year 2040	5 to 37	140 to 1100	5 to 37	140 to 1100	NA	NA

(a) NA = not applicable.

for discrimination in the selection of a disposal program. For example, it can be argued that the longer period for research and development provided by the alternative program can in turn reduce the probability of failure by producing more knowledge and a greater diversity of choice in selecting a disposal method. Such an argument has merit only if the proposed program:

- failed to maintain R&D programs in place to increase the body of knowledge
- failed to maintain a broad base of investigation of alternative media, geology and locations so as to increase the available diversity
- failed to require technical conservatism to compensate for uncertainties and adequate factors of safety
- failed to provide for reversibility of current decisions through use of concepts of retrievability or other step-wise approaches to final decisions. This reversibility allows the increased knowledge which develops over time to be a factor in near-term decisions.

To the extent that the proposed program provides for use of the above mitigating factors, it is likely that this program would achieve safety and assurance of effective permanent disposal comparable to that of the alternative program. One purpose of including the above mitigating factors would be to make it likely that the significant long-term consequences would be indistinguishable relative to an alternative strategy.

Between similar program strategies, then, the issue becomes one of degree rather than sharp difference. Do the mitigating factors adequately compensate for the existence of uncertainties? Often such questions can only be resolved by consideration of extensive detail. In such a case, one must look to the near-term aspects of the strategies, rather than to their long-term aspects in order to evaluate significant difference which can be identified with confidence.

Reviews by the Interagency Review Group (IRG) and others indicate that the R&D program must continue to obtain necessary information before proceeding with any waste isolation concept. This program of R&D is discussed in Section 5.2 and equivalent sections throughout the Statement. Longer time spent on R&D does allow the reduction of uncertainty in understanding of key processes and parameters but generally only to a certain point. Judgments need to be made as to when sufficient R&D has been conducted and information is adequate to proceed with implementing any concept. A comprehensive discussion of the resolution of uncertainties concerning geologic disposal is contained in paragraph 2D of Appendix A to the IRG Subgroup I draft report (IRG 1979). Licensing criteria and formal consideration by DOE and by independent licensing authorities through a step-wise approach will be the mechanism for making the determination of whether enough R&D has been completed.

Any repository developed after a careful siting investigation that thoroughly examines the geological considerations discussed in Section 5.2, that proceeds in a stepwise fashion of development using technically conservative placement at each step, and that is vigorously scrutinized by independent licensing authorities should not represent a substantially greater long-term risk than any other concept.

7.5 SYSTEM RESOURCE COMMITMENTS

Estimates of required commitments for major resources for construction and operation of the entire waste management system were developed for each of the nuclear growth assumptions and for each repository and reprocessing startup date. The resources considered include steel, cement, diesel fuel, gasoline, propane, electricity and manpower. The estimated resource commitments for two cases used as reference cases for resource commitments comparisons are shown in Table 7.5.1. Resource commitments for other cases are summarized here in terms of ratios to the requirements for these reference cases. A detailed listing of these resource commitments for each case can be found in Appendix A.

The reference cases in Table 7.5.1 represent resource commitments using the Case 3 growth assumptions and a 1990 repository for the once-through cycle and a 1990 reprocessing date and a 1990 repository for the reprocessing cycle. Requirements considering all four geologic media are shown. Resource commitment variations for the different geologic media are relatively small. Requirements for reprocessing are somewhat higher than for the once-through cycle in the case of steel, cement, electricity, and manpower; are about the same to somewhat higher for diesel fuel and gasoline; and are substantially higher for propane. The higher propane requirement results from incineration of combustible waste. Gasoline and diesel fuel are used primarily in transportation. These fuel requirements are based on present practice and can be expected to change as fuel-use patterns change generally. The propane requirements for the reprocessing cycle represent about 0.5% of the total U.S. consumption for the period to year 2050 assuming current consumption rates hold constant. The largest diesel fuel use amounts to about 1% of total U.S. consumption over the period. Electricity consumption amounts of 0.02 to 0.05% to the total energy generated by the nuclear power system in this case.

The resource commitments for the program alternatives using the once-through cycle are compared in Table 7.5.2 in terms of ratios relative to the quantities in Table 7.5.1. These comparisons, which are shown as ranges, take into account the range of repository startup dates considered and the four different geologic media. In general, the requirements increase with the size of the nuclear system served. With the exception of the present inventory case, which changes only slightly, requirements for the alternative program compared to the proposed program tend to range up to 2 to 3 times higher for steel, cement, gasoline, propane, and manpower and modestly higher for diesel fuel and electricity. Requirements for the no-action alternative are zero in the present inventory case and are about the same as the alternative program for steel, cement, gasoline, propane, and manpower, but diesel and electricity consumption are much lower.

Relative resource commitments for the program alternatives in the reprocessing cycle are compared in Table 7.5.3. Requirements for the alternative program tend to be about the same to somewhat higher than the proposed program requirements.

TABLE 7.5.1 Resource Commitment Reference Cases^(a)

<u>Salt</u>	<u>Once-Through Cycle, 1990 Repository</u>	<u>Reprocessing Cycle, 1990 Reprocessing and 1990 Repository</u>
Steel, MT	3.0×10^5	4.8×10^5
Cement, MT	2.8×10^5	5.5×10^5
Diesel Fuel, m ³	1.6×10^6	1.4×10^6
Gasoline, m ³	7.9×10^4	1.1×10^5
Propane, m ³	1.1×10^4	3.5×10^7
Electricity, kWh	6.1×10^9	1.8×10^{10}
Man Power, man-yr	8.9×10^4	1.4×10^5
<u>Granite</u>		
Steel, MT	4.9×10^5	7.2×10^5
Cement, MT	3.0×10^5	6.2×10^5
Diesel Fuel, m ³	1.4×10^6	1.4×10^6
Gasoline, m ³	8.6×10^4	1.5×10^5
Propane, m ³	1.3×10^4	3.5×10^7
Electricity, kWh	5.8×10^9	1.9×10^{10}
Man Power, man-yr	9.4×10^4	1.8×10^5
<u>Shale</u>		
Steel, MT	2.9×10^5	3.8×10^5
Cement, MT	2.9×10^5	6.4×10^5
Diesel Fuel, m ³	1.5×10^6	1.6×10^6
Gasoline, m ³	7.5×10^4	1.5×10^5
Propane, m ³	1.2×10^4	3.5×10^7
Electricity, kWh	5.4×10^9	1.9×10^{10}
Man Power, man-yr	8.6×10^4	1.8×10^5
<u>Basalt</u>		
Steel, MT	4.8×10^5	7.4×10^5
Cement, MT	2.7×10^5	6.1×10^5
Diesel Fuel, m ³	1.4×10^6	1.4×10^6
Gasoline, m ³	7.8×10^4	1.5×10^5
Propane, m ³	1.1×10^4	3.5×10^7
Electricity, kWh	5.8×10^9	1.8×10^{10}
Man Power, man-yr	9.9×10^4	2.0×10^5

(a) Case 3 growth assumption with 1990 repositories and 1990 reprocessing.

TABLE 7.5.2 Comparison of Relative Resource Commitments for the Program Alternatives Using the Once-Through Fuel Cycle(a)

Case	Nuclear Power Growth Assumption	Steel, MT	Cement, MT	Proposed Program			Electricity, kWh	Man-Power, man-yr
				Diesel, m ³	Gasoline, m ³	Propane, m ³		
1	Present Inventory Only	.02 to .05	.01 to .02	.03 to .04	.02 to .03	.02 to .03	.03	.03 to .04
2	Present Capacity with Normal Life	.29 to .77	.43 to 1.5	.18 to .26	.03 to .07	.28 to .69	.18 to .26	.26 to .48
3	250 GWe by Year 2000 with Normal Life	.97 to 3.3	.96 to 5.7	.88 to 1.2	.95 to 2.7	1.0 to 2.7	.89 to 1.2	.97 to 2.0
4	250 GWe by Year 2000 and Steady State to 2040	2.1 to 3.0	3.3 to 3.4	1.2 to 1.4	2.0 to 2.2	2.1 to 2.2	1.2 to 1.5	1.7 to 1.9
5	500 GWe System by Year 2040	2.5 to 3.7	3.6	1.6 to 1.9	2.3 to 2.5	2.5 to 2.6	1.6 to 2.0	2.2 to 2.4
<u>Alternative Program</u>								
1	Present Inventory Only	.02 to .03	.008 to .02	.03 to .04	.02 to .03	.02 to .03	.02 to .03	.02 to .03
2	Present Capacity with Normal Life	.70 to .90	1.5 to 1.8	.20 to .26	.06 to .76	.62 to .80	.20 to .28	.42 to .55
3	250 GWe by Year 2000 with Normal Life	2.9 to 4.3	5.7 to 8.9	.94 to 1.3	2.4 to 3.7	2.5 to 3.9	1.0 to 1.3	1.9 to 2.7
4	250 GWe by Year 2000 and Steady State to 2040	4.0 to 4.7	8.6	1.4 to 1.7	3.7 to 3.8	3.8 to 4.1	1.3 to 1.6	2.7 to 3.0
5	500 GWe System by Year 2040	5.3 to 6.0	11.1	1.8 to 2.3	4.6 to 4.9	4.8 to 5.2	1.8 to 2.3	3.5 to 3.8
<u>No-Action Alternative</u>								
1	Present Inventory Only	0	0	0	0	0	0	0
2	Present Capacity with Normal Life	.70	1.8	.07	.59	.64	.09	.36
3	250 GWe by Year 2000 with Normal Life	3.7	9.3	.04	3.2	3.4	.46	1.9

(a) Case 3 with a 1990 repository in salt was used as the reference for these ratios.

TABLE 7.5.3 Comparison of Relative Resource Commitments for the Program Alternatives Using the Reprocessing Cycle(a)

Case	Nuclear Power Growth Assumption	Steel, MT	Cement, MT	Proposed Program			Electricity, kWh	Man-Power, man-yr
				Diesel, m ³	Gasoline, m ³	Propane, m ³		
3	250 GWe by Year 2000 With Normal Life	.97 to 2.3	.96 to 3.5	.88 to 1.1	.57 to 2.3	.97 to 1.3	.89 to 1.0	.97 to 1.7
4	250 GWe by Year 2000 and Steady State to 2040	1.5 to 2.3	2.4 to 2.7	1.2 to 1.4	1.6 to 2.2	1.3	1.3	1.5 to 1.9
5	500 GWe System by Year 2040	1.7 to 2.9	2.5 to 2.9	1.6 to 2.0	2.1 to 2.7	1.7	1.7 to 1.8	1.9 to 2.5
<u>Alternative Program</u>								
3	250 GWe by Year 2000 With Normal Life	1.2 to 2.9	1.1 to 3.6	.93 to 1.4	.57 to 3.5	.97 to 1.0	.94 to 1.0	1.1 to 1.9
4	250 GWe by Year 2000 And Steady State to 2040	2.0 to 2.9	2.7 to 2.9	1.6 to 1.9	2.8 to 3.3	1.3	1.3	1.6 to 2.0
5	500 GWe System by Year 2040	2.5 to 3.5	3.1 to 3.5	2.2 to 2.5	3.5 to 4.1	1.7	1.7 to 1.8	2.1 to 2.6

(a) Case 3 with 1990 reprocessing and a 1990 repository in salt was used as the reference for these ratios.

7.6 SYSTEM COSTS

Costs for the entire waste management system are presented in this section. The costs include all predisposal and disposal costs from reactor discharge of the spent fuel to final isolation of the waste in a disposal repository. The wastes include spent fuel in the once-through cycle and high-level and TRU wastes in the reprocessing cycle. The costs include the estimated expenditures by the Federal Government for research and development and repository multiple-site qualification.^(a) It is assumed that these R&D costs will be recovered in accordance with the President's February 12, 1980 statement, "through fees paid by the utilities" for storage at government-owned storage facilities and for disposal at the final disposal repositories. Costs are presented here both in terms of total dollars and in terms of mills/kWh, so that the impact of this waste management on nuclear power costs can be put into perspective.

One of the most important cost components of the waste management systems is the Department of Energy's research and development and site qualification cost. The estimated annual R&D expenditures through 1995 for predisposal management of commercial wastes are tabulated in Appendix Table A.9.5. The estimated annual expenditures for disposal R&D and repository site qualification work are tabulated in Appendix Table A.9.6. Separate schedules are shown for each repository startup date considered in this analysis. The total estimated R&D and multiple site qualification costs are summarized in Table 7.6.1. These costs also include cumulative expenditures through 1980.

TABLE 7.6.1 Total Estimated Research and Development and Multiple Site Qualification Costs, \$ millions

<u>Case</u>	<u>Date of First Repository</u>	<u>Total Predisposal R&D</u>	<u>Total Disposal R&D and Site Verification</u>	<u>Total</u>
1 & 2	1990	500	2,400	2,900
	2010	800	3,200	4,000
	2030	900	8,000	8,900
3, 4 & 5	1990	600	3,000	3,600
	2000	800	3,200	4,000
	2010	900	3,700	4,600
	2020	1,000	6,600	7,600
	2030	1,000	8,500	9,500

The R&D and multiple site qualification costs for the year 2000 repository represent an estimate for DOE's present program plan and are consistent with the program description and schedule of activities outlined in DOE's Confidence Rulemaking Statement (DOE/NE-0007

(a) "When four or five sites have been evaluated and found potentially suitable, one or more will be selected for further development as a licensed full-scale repository." President Carter, Feb. 12, 1980."

1980). (This schedule actually leads to a first repository in 1997, so some of the expenditures occur a little earlier than would be the case for a year 2000 startup.) For the 1990 repository opening, costs for activities that could not be completed by that time are deleted. Second and third repositories in 1995 and 2000 are assumed. For the 2010 repository opening, it has been assumed that the delay is caused in half by political, regulatory or other reasons at no cost and in half by technical problems with siting, licensing or other factors. Second and third repositories in 2015 and 2020 are assumed. For the 2020 and 2030 repository openings (dates within the alternative program envelope), it was assumed that expenditures continue at the 1981 level (\$190 million/yr) with the program restructured to give equal emphasis to two or three disposal technologies. At the year 2000 and 2010, respectively, a preferred technology is selected and the expenditure rate is reduced by one-third. After the first repository opening (2020 and 2030, respectively), the expenditure rate is halved and continues for another 10 years when R&D is assumed to be completed.

For Cases 1 and 2, where only one repository is required, the R&D and multiple site qualification costs are reduced and phased out earlier. For the "no-action" alternative cases only the costs of R&D expended through 1980 plus the spent fuel storage R&D costs (Table A.9.5) are included, for a total of \$614 million.

The total waste management costs in billions of dollars are compared for the program alternatives when using the once-through cycle in Table 7.6.2 and in Table 7.6.3 when using the reprocessing cycle. The range of costs takes into account the variation of costs with disposal and reprocessing dates and the variation in costs with the four disposal media that were considered and include the estimated R&D multiple site qualification costs. The costs increase as one would expect with the higher nuclear growth assumptions. However, they are disproportionately high for the very low growth assumptions because of the fixed costs for facilities and research and development costs. For the three cases where the no-action alternative was evaluated, the costs are similar to the low end to mid-range of the range for the proposed program. With the once-through cycle, the cost ranges are significantly higher for

TABLE 7.6.2. Comparison of Total Waste Management Costs for the Program Alternatives Using the Once-Through Cycle, \$ Billions

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)	No-Action Alternative
1	Present Inventory Only	5.1 to 7.6	7.4 to 14	6.4
2	Present Capacity and Normal Life	11 to 18	16 to 24	12
3	250 GWe System by Year 2000 and Normal Life	39 to 68	60 to 82	49
4	250 GWe System by Year 2000 and Steady State	61 to 72	87 to 98	NA(a)
5	500 GWe System by Year 2040	78 to 93	116 to 131	NA

(a) NA = not applicable.

TABLE 7.6.3. Comparison of Total Waste Management Costs for the Program Alternatives Using the Reprocessing Cycle,^(a) \$ Billions

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)	No-Action Alternative
1	Present Inventory Only	NA ^(b)	NA	NA
2	Present Capacity and Normal Life	NA	NA	NA
3	250 GWe System by Year 2000 and Normal Life	59 to 90	58 to 90	NA
4	250 GWe System by Year 2000 and Steady State	87 to 108	89 to 104	NA
5	500 GWe System by Year 2040	114 to 146	116 to 137	NA

(a) Assumed reprocessing startup dates range from 1990 to 2010.

(b) NA = not applicable.

the alternative program than for the proposed or no-action alternatives. With the reprocessing cycle, the cost ranges are about the same for both the proposed and alternative programs.

Costs for the program alternative are compared on the basis of levelized unit costs in terms of mills/kWh at a 0% discount rate in Table 7.6.4 for the once-through cycle and Table 7.6.5 for the reprocessing cycle. On this basis, unit cost ranges for the present inventory case (Case 1) are much higher than the other cases because of the small quantity of kilowatt-hours generated in this case relative to the fixed costs. With the present capacity case (Case 2), the costs drop to about 1/3 of the Case 1 costs. For the once-through cycle, the alternative program unit costs range higher than the proposed program and the no-action alternative costs lie at the low end to mid-range of the proposed program cost range. Costs are higher for the proposed program using the reprocessing cycle than are the costs of the once-through cycle, but the cost range for the alternative program is almost identical to the proposed program range.

When a discount rate larger than zero is used to calculate levelized costs, the differences between the proposed program and the alternative program and differences between once-through and reprocessing cycles become less pronounced. This is shown in Tables 7.6.6 and 7.6.7, which compare the costs for the once-through cycle and the reprocessing cycle on the basis of a 7% discount rate and in Tables 7.6.8 and 7.6.9, which compare the same cost ranges on the basis of a 10% discount rate.

At a 7% discount rate, cost differences between the proposed program and the alternative program are not significant for either the once-through cycle or the reprocessing cycle. Costs for the reprocessing cycle range mostly about 10% higher to as much as 30% higher than for the once-through cycle.

At a 10% discount rate, as with a 7% rate, the cost differences between the proposed program and the alternative program are not significant. The costs for the reprocessing cycle range from slightly higher to as much as 15% higher than for the once-through cycle.

TABLE 7.6.4. Comparison of Levelized Waste-Management Unit Costs for the Program Alternatives Using the Once-Through Cycle and a 0% Discount Rate, mills/kWh^(a)

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)	No-Action Alternative
1	Present Inventory Only	2.9 to 4.3	4.2 to 7.7	3.6
2	Present Capacity and Normal Life	1.0 to 1.6	1.5 to 2.2	1.1
3	250 GWe System by year 2000 and Normal Life	0.7 to 1.2	1.1 to 1.5	0.9
4	250 GWe System by Year 2000 and Steady State	0.8 to 1.0	1.1 to 1.3	NA ^(b)
5	500 GWe System by Year 2040	0.7 to 0.9	1.1 to 1.2	NA

(a) To convert mills/kWh to \$/kg HM multiply by 233.

(b) NA = not applicable.

TABLE 7.6.5. Comparison of Levelized Waste-Management Unit Costs for the Program Alternatives Using the Reprocessing Cycle and a 0% Discount Rate, mills/kWh

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)	No-Action Alternative
1	Present Inventory Only	NA ^(a)	NA	NA
2	Present Capacity and Normal Life	NA	NA	NA
3	250 GWe System by Year 2000 and Normal Life	1.0 to 1.6	1.0 to 1.6	NA
4	250 GWe System by Year 2000 and Steady State	1.1 to 1.4	1.2 to 1.4	NA
5	500 GWe System by Year 2040	1.1 to 1.4	1.1 to 1.3	NA

(a) NA = not applicable.

A series of tables in Appendix A (Tables A.9.3a to A.9.4c) present total unit costs for each of the four geologic media over the range of 0 to 10% discount rates. These tabulations indicate generally small variations in total unit costs with the different repository media. The largest differences show up in the reprocessing cycle with early reprocessing.

Another series of tables in Appendix A (Tables A.9.1a to A.9.2c) show a breakdown of the total unit costs between spent fuel storage and transport, spent fuel treatment, other waste treatment storage and transport, disposal, and research and development. These tables show that for the once-through cycle, the research and development and site qualification cost is the dominant cost over the entire range of discount rates in the present inventory case. For the higher nuclear growth cases (cases 3, 4 and 5), research and development costs are less than 10% of the total costs at a 0% discount rate but account for one-third to one-half the cost at a 10% discount rate. Disposal costs tend to become a smaller

TABLE 7.6.6. Comparison of Levelized Waste-Management Unit Costs for the Program Alternatives Using the Once-Through Cycle and a 7% Discount Rate, mills/kWh

<u>Case</u>	<u>Nuclear Power Growth Assumption</u>	<u>Proposed Program (Geologic Disposal Starting 1990 - 2010)</u>	<u>Alternative Program (Disposal Starting 2010 - 2030)</u>	<u>No-Action Alternative</u>
1	Present Inventory Only	1.6 to 1.7	1.6 to 2.0	0.78
2	Present Capacity and Normal Life	0.85 to 0.92	0.87 to 1.00	0.56
3	250 GWe system by Year 2000 and Normal Life	0.61 to 0.69	0.65 to 0.68	0.49
4	250 GWe System by Year 2000 and Steady State	0.66 to 0.71	0.67 to 0.69	NA(a)
5	500 GWe System by Year 2040	0.64 to 0.69	0.66 to 0.67	NA

(a) NA = not applicable.

TABLE 7.6.7. Comparison of Levelized Waste-Management Unit Costs for the Program Alternatives Using the Reprocessing Cycle and a 7% Discount Rate, mills/kWh

<u>Case</u>	<u>Nuclear Power Growth Assumption</u>	<u>Proposed Program (Geologic Disposal Starting 1990 - 2010)</u>	<u>Alternative Program (Disposal Starting 2010 - 2030)</u>	<u>No-Action Alternative</u>
1	Present Inventory Only	NA(a)	NA	NA
2	Present Capacity and Normal Life	NA	NA	NA
3	250 Gwe System by Year 2000 and Normal Life	0.68 to 0.91	0.68 to 0.72	NA
4	250 GWe system by Year 2000 and Steady State	0.73 to 0.79	0.73 to 0.74	NA
5	500 GWe System by Year 2040	0.72 to 0.79	0.71 to 0.73	NA

(a) NA = not applicable.

portion of the total as the discount rate increases because they are incurred a number of years after the power is generated and thus are discounted proportionately more. In the reprocessing cycle, the research and development costs also, as in the once-through cycle, increase in importance as the discount rate is increased. Waste treatment and storage costs drop off significantly as the discount rate increases because these costs are deferred relative to the time of power generation. In both cycles, although spent-fuel storage and transport costs decline as the discount rate increases, they always remain a substantial portion of the total cost because they are incurred relatively soon after discharge and thus are not as heavily discounted as some of the other costs. For example, in the reprocessing cycle, spent-fuel storage and transport costs account for 30 to 60% of the total costs at a 10% discount rate compared to 20 to 50% at a 0% discount rate.

Although the total expenditure for waste management is quite large, it does not, except for the present inventory case, add more than 2 to 10%, and most likely not more than 3%, to

TABLE 7.6.8. Comparison of Levelized Waste-Management Costs for the Program Alternatives Using the Once-Through Cycle and a 10% Discount Rate, mills/kWh

<u>Case</u>	<u>Nuclear Power Growth Assumption</u>	<u>Proposed Program (Geologic Disposal Starting 1990 - 2010)</u>	<u>Alternative Program (Disposal Starting 2010 - 2030)</u>	<u>No-Action Alternative</u>
1	Present Inventory Only	1.2 to 1.4	1.2 to 1.4	0.61
2	Present Capacity and Normal Life	0.77 to 0.83	0.77 to 0.85	0.50
3	250 GWe system by Year 2000 and Normal Life	0.58 to 0.65	0.58 to 0.61	0.44
4	250 GWe System by Year 2000 and Steady State	0.61 to 0.63	0.60 to 0.61	NA(a)
5	500 GWe System by Year 2040	0.60 to 0.62	0.59 to 0.60	NA

(a) NA = not available.

TABLE 7.6.9. Comparison of Levelized Waste-Management Costs for the Program Alternatives Using the Reprocessing Cycle and a 10% Discount Rate, mills/kWh

<u>Case</u>	<u>Nuclear Power Growth Assumption</u>	<u>Proposed Program (Geologic Disposal Starting 1990 - 2010)</u>	<u>Alternative Program (Disposal Starting 2010 - 2030)</u>	<u>No-Action Alternative</u>
1	Present Inventory Only	NA(a)	NA	NA
2	Present Capacity and Normal Life	NA	NA	NA
3	250 Gwe System by Year 2000 and Normal Life	0.59 to 0.77	0.59 to 0.63	NA
4	250 GWe system by Year 2000 and Steady State	0.63 to 0.66	0.63 to 0.64	NA
5	500 GWe System by Year 2040	0.63 to 0.66	0.62 to 0.63	NA

(a) NA = not available.

the total cost of nuclear power generation, which is estimated in terms of 1978 dollars to range from 25 to 35 mills/kWh for a new facility. It is also of interest to note that although the estimated expenditures for R&D and repository site qualification are very large, they amount to less than 0.5 mills/kWh (except in the present inventory case when it amounts to 2 to 5 mills/kWh at a 0% discount rate) when allocated to the generated electrical energy.

7.7 SYSTEM SIMULATION CONCLUSIONS

The system simulation analysis shows that the environmental impact of high-level and TRU waste management will be only slightly affected by waste management programs and the program strategy selected by DOE. More specifically, regarding the three program alternatives considered in this statement, the following conclusions can be drawn:

1. Radiation dose accumulations for normal operation of the required facilities increase as the size of the nuclear system increase. Neither the dose accumulation nor health effects are significantly different for the program alternatives in either the once-through or reprocessing cycles. The dose accumulation with spent fuel reprocessing is 0.5% of the regional and 0.003% of the worldwide dose from natural causes over the same period.

For the once-through cycle, assuming continued nuclear growth, the regional 70-year whole body radiation dose accumulation over the period considered here lies in the range of 1,000 to 2,000 man-rem; an additional 400 to 1,000 man-rem are estimated for the worldwide accumulation. Comparable dose accumulations for the reprocessing cycle range from 13,000 to 46,000 man-rem for a region and 570,000 to 1,400,000 man-rem worldwide.

2. Resource commitments also increase with increasing size of the nuclear system. With the once-through cycle, resource requirements for the alternative program range up to 2 to 3 times higher than for the proposed program. With the reprocessing cycle, resource requirements for the alternative program are about the same to slightly higher than for the proposed program. Resource commitment variations relative to different geologic media are relatively small. Requirements for reprocessing are somewhat higher than for the once-through cycle for steel, cement, electricity, and manpower; about the same to somewhat higher for diesel fuel and gasoline; and substantially higher for propane. For all cases, resource requirements are a small fraction of current U.S. consumption rates.
3. Waste management costs increase with increasing size of the nuclear system but unit costs are disproportionately high for the very low-growth cases. With the once-through cycle, the cost range is significantly higher for the alternative program than for the proposed program. With the reprocessing cycle, the cost ranges are about the same for both alternatives. The no-action alternative costs are similar to the low end of the cost range for the proposed program with the once-through cycle.

Levelized unit costs in terms of mills/kWh are sensitive to the discount rate. At a 0% discount rate, the alternative program costs are significantly higher than the proposed program costs for the once-through cycle but are about the same for the reprocessing cycle. Costs for the reprocessing cycle are higher than costs for the once-through cycle. At discount rates in the range of 7 to 10%, the differences between the proposed and alternative programs and between the once-through and reprocessing cycles become insignificant.

Unit costs for the present inventory and present capacity cases are substantially higher than for the higher nuclear growth cases because of the small amount of electricity generated relative to the fixed costs.

Assuming a 7% discount rate and continued growth of the nuclear industry, total high-level and TRU waste management costs lie in the range of 0.6 to 1.0 mill/kWh.

4. Interim storage requirements for spent fuel are substantially greater for the alternative program than for the proposed program with the once-through cycle. With the reprocessing cycle, spent fuel storage requirements are controlled by reprocessing capacity and are not sensitive to the waste management program alternatives. Storage requirements for reprocessing waste, however, become substantial with the alternative program.

Spent fuel storage requirements are maximized with the no-action alternative.

5. Transportation requirements are higher for the alternative program compared to the proposed program with both the once-through and the reprocessing fuel cycles.

Transportation requirements are minimized with the no-action alternative.

6. Age of the waste. A potentially beneficial aspect of the alternative program is the aging of the waste, which results in reduced radioactivity and heat generation rates which can be used to reduce repository space requirements or to further reduce the temperatures in the repository.

7. Geologic repository requirements are sensitive to the geologic medium selected, the nuclear growth rate, and the fuel cycle employed. For the highest growth assumption considered here, these requirements for operations through the year 2040 range from two to seven 800-hectare repositories for the once-through cycle and from four to nine 800-hectare repositories for the reprocessing cycle.

8. The radioactivity inventory in disposal repositories is proportional to the nuclear energy generated. The ultimate accumulation is not sensitive to the time when disposal commences but is affected by the amount of plutonium recycled and thus to the time when recycle is started.

The inventory of fission and activation products is closely similar for both the once-through and reprocessing cycles. However, the actinide radioactivity inventory is larger for the once-through cycle than for the reprocessing cycle because all of the plutonium remains with the spent fuel. The difference in actinide activity between the two cycles is not, however, proportional to the amount of plutonium in the waste. This is because recycle of plutonium produces more of the higher actinides (e.g., americium and curium isotopes), which are discarded in the wastes. Thus, rather than a factor of 100, which could be expected on the basis of the amount of plutonium discarded, the actinide activity in the spent fuel waste is on the order of only 2 to 10 times larger than the reprocessing cycle wastes.

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CHAPTER 8

GLOSSARY OF KEY TERMS AND ACRONYMS

Abiotic: characterized by the absence of life.

Abyssal Hill: relatively small topographic feature of the deep ocean floor ranging to 600 to 900 m high and a few kilometers wide.

Actinides: Radioactive elements with atomic number larger than 88.

Activation: The process of making a material radioactive by bombardment with neutrons, protons, or other nuclear particles.

Activity: A measure of the rate at which radioactive material is emitting radiation; usually given in terms of the number of nuclear disintegrations occurring in a given quantity of material over a unit of time. The special unit of activity is the curie (Ci).

AFR: Away-from-reactor (spent fuel storage concept).

Aging: Usually refers to time to permit decay of short-lived radionuclides.

ALAP: As low as practicable, now generally replaced with ALARA (as low as reasonably achievable).

ALARA: As low as reasonably achievable. ALARA refers to limiting release and exposure and is used by the NRC (10 CFR 50.34) in the context of ". . . as low as reasonably achievable taking into account the state of technology, and the economics of improvements in relation to benefits to the public health and safety and other societal and socioeconomic considerations. . ."

Allowance Item: A number, arrived at by judgement, that represents material or equipment cost that cannot be developed otherwise because of the absence of design detail.

Alluvial Fan: A sloping, fan-shaped mass of loose rock material deposited by a stream at the place where it emerges from an upland onto a broad valley or a plain.

Alluvium: All detrital material deposited permanently or in transit by streams.

Alpha Particle: A positively charged particle emitted by certain radioactive material. It is made up of two neutrons and two protons; hence it is identical with the nucleus of a helium atom.

Amphibole: A group of dark, rock-forming, ferromagnesian silicate minerals which are closely related in crystal form and composition and which have abundant and wide distribution in igneous and metamorphic rocks.

Andesitic: A volcanic rock composed primarily of the plagioclase feldspar andesine and one or more mafic constituents.

Anion: An ion that is negatively charged.

Anticline: A fold, the core of which contains stratigraphically older rocks, which in simplest form is elongate and convex upward with the two limbs dipping away from each other.

APS: Atmospheric protection system.

Aquifer: A water-bearing layer of permeable rock or soil that will yield water in usable quantities to wells.

Aquitard: A natural rock or soil of low permeability which is stratigraphically adjacent to one or more aquifers and through which water movement is markedly retarded or impeded.

Argillaceous: Containing or pertaining to clay.

Artesian: When pertaining to an aquifer, it is one that is confined so that its hydraulic head rises above the top of the aquifer unit; thus an artesian water body is one that is confined under hydraulic pressure.

Atom: An electrically neutral particle of matter, indivisible by chemical means.

Atomic Number: The number of protons within an atomic nucleus.

Atomic Weight: The mass of an atom relative to other atoms.

Back End of the Fuel Cycle: Includes spent fuel storage, fuel reprocessing, mixed-oxide fuel fabrication, and waste management.

Background Radiation: The radiation in man's natural and undisturbed environment. It results from cosmic rays and from the naturally radioactive elements of the earth, including those from within the human body.

Basement Rock: A complex of undifferentiated rocks that underlies the oldest identifiable rocks in the area.

Basin: A depressed area generally having no outlet for surface water.

Batholith: A shield-shaped mass of igneous-intruded rock, greater than 100 km² in area, extending to great depth and whose diameter increases with depth.

Bedrock: A solid rock formation usually underlying one or more other loose formations.

Benthic: Refers to the bottom of a body of water.

Bentonitic: Pertaining to rock containing bentonite, a clay formed from the decomposition of volcanic ash.

Biosphere: The part of the earth in which life can exist, including the lithosphere, hydrosphere, and atmosphere; living beings together with their environment.

Biota: The animal and plant life of a region.

Biotite: A complex silicate of aluminum, potassium, magnesium, and iron with hydroxyl that is a widely distributed and important rock-forming mineral of the mica group.

Block-Faulting: A type of vertical faulting in which the crust is divided into structural or fault blocks of different elevations and orientations.

Boiling Water Reactor (BWR): A reactor system that uses a boiling water primary cooling system. Primary cooling system steam turns turbines to generate electricity.

Borosilicate Glass: A silicate glass containing at least 5 percent boric acid and used to vitrify calcined waste.

Breccia: A coarse-grained clastic rock composed of large, angular, and broken rock fragments cemented together in a finer grained matrix.

Burial Grounds: Areas designated for disposal of containers of radioactive wastes and obsolete or worn-out equipment by near-surface burial.

Calcine: Material heated to a temperature below its melting point to bring about loss of moisture and oxidation.

Canister: A metal container for radioactive solid waste.

Case: A container that provides shielding and containment during transportation of radioactive materials.

Catastrophic: A violent, sudden or unexpected event which results in failure of the predicted performance of a system or component.

Cation: An ion that is positively charged.

Cation Exchange Chromatography (CEC): A process for separating several cations using the differences in the rate they travel on an ion exchange column.

Cermet: A material made by combining a heat resistant ceramic with a metal usually made by powder metallurgy.

CH-TRU: Contact-handled TRU waste.

Clastic: Pertaining to or the state of being a rock or sediment composed principally of broken fragments derived from preexisting rocks or minerals.

Colocated: Refers to location of facilities at a common site.

Concentration Guide: The average concentration of a radionuclide in air or water to which a worker or member of the general public may be continuously exposed without exceeding radiation dose standards.

Consolidated (material): In geology, natural materials that have been made firm, cohesive, and hard.

Contact-Handled Waste: Waste package having surface dose rate less than 0.2 R/hr. Such packages can be handled by workers without extensive shielding. Contact-handled wastes were termed low-level wastes in DOE/ET-0028 and DOE/ET-0029.

Containment: Confining the radioactive wastes within presented boundaries, e.g., within a waste package.

Contingency (cost): The amount of money added to the estimated cost of a project to cover certain areas of cost uncertainty and reduce the probability of understating the project cost estimate. With the contingency added, there is a more nearly equal probability of a cost underrun or overrun.

Cost of Money: Weighted cost of debt and equity financing. Cost of money is used synonymously with cost of capital.

Critical Mass: The mass of fissionable material of a particular shape that is just sufficient to sustain a nuclear chain reaction.

Criticality: The condition in which a nuclear reactor is just self-sustaining.

Crystalline Rock: An inexact but convenient term designating an igneous or metamorphic rock, as opposed to a sedimentary rock.

Curie (Ci): A special unit of activity where 1 Ci equals 3.7×10^{10} spontaneous nuclear disintegrations per second.

Daughter Nuclide: A nuclide formed upon disintegration of a parent radionuclide.

Decommissioning: Preparations taken for retirement from active service of nuclear facilities, accompanied by the execution of a program to reduce or stabilize radioactive contamination. The objective of decommissioning is to place the facility in such a condition that future risk to public safety from the facility is within acceptable bounds.

Decontamination: The selective removal of radioactive material from a surface or from within another material.

- Decontamination Factor (DF):** The ratio of the original contamination level to the contamination level after decontamination.
- Deep Continental Geologic Formations:** Geologic media beneath the continents and isolated from the land surface by several hundred to thousands of meters of overlying rock material.
- Depositional Environment (sedimentary environment):** A geographically restricted environment where sediment accumulates under similar physical, chemical, and biological conditions.
- Devitrification:** The process by which glassy substances lose their vitreous nature and become crystalline.
- Diaprisim:** The piercing of overlying rocks by an upward-moving mobile core or material, such as a salt body or an igneous intrusion.
- Discharge:** In ground-water hydrology, water that issues naturally or is withdrawn from an aquifer.
- Disposal (radioactive waste):** The planned release of radioactive waste in a manner which is considered permanent so that recovery is not provided for.
- Dome:** A dome-shaped landform or rock mass; a large igneous intrusion whose surface is convex upward with sides sloping away at low but gradually increasing angles; an uplift or an anticlinal structure, either circular or elliptical in outline, in which the rock dips gently away in all directions, for example, a salt dome.
- Dissolution:** In this context it refers to the dissolving of spent fuel by nitric acid as a process step in fuel reprocessing.
- Dose:** Herein generally means the more rigorous term "dose-equivalent." The latter, expressed in units of rem, implies a consistent basis for estimates of consequential health risk, regardless of rate, quantity, source, or quality of the radiation exposure.
- DOT:** U.S. Department of Transportation.
- Dry Storage:** Storage of waste packages without liquid cooling.
- EIA:** Energy Information Administration.
- EPA:** U.S. Environmental Protection Agency.
- Epeirogeny:** The broad movements of uplift and subsidence which affect whole or large portions of continents or ocean basins.
- Fault:** A fracture or fracture zone along which there has been displacement of the sides relative to one another parallel to the fracture.
- Fault Block:** A crustal unit either completely or partly bounded by faults.
- Fault System:** A system of parallel or nearly parallel faults that are related to a particular deformational episode.
- Feldspar:** Any of a group of common rock-forming minerals that are silicates of alumina and some other base, such as potash, soda, or lime.
- Fission (nuclear):** The splitting of a nucleus into two or (rarely) more fragments; usually limited to heavier nuclei such as isotopes of uranium, plutonium, and thorium.
- Fission Product:** Any radioactive or stable nuclide produced by fission, including both primary fission fragments and their radioactive decay products.
- Fissionable Material:** Actinides capable of undergoing fission by interaction with neutrons of all energies.

FPF: Fuel packaging facility.

Fracture: breaks in rocks caused by intense folding or faulting or the process of breaking fluid-bearing strata by injecting a fluid under such pressure as to cause partings in the rock.

Freshwater Lens: A body of fresh water roughly shaped like a lens formed as a result of injecting freshwater into a salt water body or occurring naturally when precipitation infiltrates a saline aquifer.

Fuel (nuclear reactor): Fissionable material used as the source of power when placed in a critical arrangement in a nuclear reactor.

Fuel Cycle: Mining, refining, enrichment, and fabrication of fuel elements, use in a reactor, chemical processing to recover the fissionable material remaining in the spent fuel, reenrichment of the fuel material, refabrication of new fuel elements, and management of radioactive waste.

Fuel Element: A tube, rod, or other form into which fissionable material is fabricated for use in a reactor.

Fuel Reprocessing Plant (FRP): Plant where irradiated fuel elements are dissolved, waste materials removed, and reusable materials are segregated for reuse.

Fuel Residue Waste (FRW): Solid wastes consisting of the residue (fuel element hardware and chopped cladding material) after the bulk of fuel core material, including most of the actinides and fission products, has been dissolved in nitric acid.

Gamma Ray: Electromagnetic radiation, similar in nature to x-rays, emitted by the nuclei of some radioactive substances during radioactive decay.

GEIS: Generic Environmental Impact Statement.

Geohydrology: The study of the character, source, and mode of occurrence of underground water.

Geothermal: Pertaining to the heat of the interior of the earth.

Geothermal Gradient: The increasing temperature of the earth with depth.

GESMO: Generic Environmental Impact Statement on use of Mixed-Oxide fuel in LWRs.

Granitic: Of or pertaining to granite. Granite-like.

Granitoid: A textural term indicating grain size and mineral distribution typical of granite.

Ground Water: Water that exists or flows within the zone of saturation beneath the land surface.

Grout: A mortar fluid combined with liquid waste to provide a matrix for isolation of the waste and to seal the waste from the environment.

GWe: Gigawatts (billions of watts) of electrical generation; a rate of energy production.

Gyre: A large closed ringlike system of ocean currents which rotates in a circular motion in each of the major ocean basins.

Half-Life: a) physical--the time required for quantity of a radioactive substance to decay to one-half of its original quantity. b) biological--time required for half of an ingested or inhaled substance to be eliminated from the body by natural process. c) effective--time required for half of an ingested or inhaled radioactive substance to be eliminated from the body by the combination of radioactive decay and natural processes; mathematically equal to product of the physical and biological half-lives divided by the sum of the physical and biological half-lives.

Head End of the Fuel Cycle: Mining, milling, enrichment, and fabrication of UO₂ fuel.

HEPA: High-efficiency particulate air (filter).

High-Level Liquid Waste (HLLW): The aqueous waste resulting from operation of the first cycle solvent extraction system (or its equivalent) in a facility for reprocessing irradiated reactor fuels as well as concentrated wastes from subsequent cycles.

High-Level Waste (HLW): DOE management directives define high-level waste to include high-level liquid wastes, products from solidification of high-level liquid waste, and irradiated fuel elements if discarded without reprocessing. A proposed NRC regulation (10 CFR 60.3) defines high-level waste to include irradiated fuel, high-level liquid waste, and products from its solidification. In the GEIS there are instances, however, where discarded spent fuel and high-level waste (as wastes from the reprocessing of spent fuel) are cited separately.

Highly Enriched Uranium (HEU): Uranium containing 5% or more of added ²³⁵U.

HM: Heavy metal, generally uranium and plutonium.

Hornblende: A common member of the amphibole group of minerals.

Hot Cell: A facility which allows remote viewing and manipulation of radioactive substances.

Hydraulic Gradient: The change in static head per unit of lateral distance in a given direction.

Hydrologic: Pertaining to the study of the properties, distribution and circulation of water on the surface of the land, in the soil and underlying rocks, and in the atmosphere.

Hydrostatic Pressure: The pressure exerted by the water at any given point in a body of water at rest.

ICPP: Idaho Chemical Processing Plant.

ICRP: International Commission on Radiological Protection.

ILLW: Intermediate-level liquid waste.

Immobilization: Treatment and/or emplacement of the wastes so as to impede their movement.

Interim Storage: Storage operations for which a) monitoring and human control are provided and b) subsequent action involving treatment, transportation, or final disposition is expected.

Interstices: In geology, small openings between solid particles in a rock or unconsolidated material; may be a void or pore and often contains ground water. Interstitial permeability is used to differentiate interconnected pore permeability from fracture permeability.

Ion Exchange: Replacement of ions adsorbed on a solid, such as a clay particle, or exposed at the surface of a solid by ions from solution, usually in natural water. The phenomenon is known to occur when natural water moves through clays, zeolitic rocks, and other materials of the earth's crust.

ISFS: Independent spent fuel storage.

ISFSF: Independent spent fuel storage facility.

Isolation: Segregating wastes from the accessible environment (biosphere) to the extent required to meet applicable radiological performance objectives.

Joint: A fracture or parting in a rock, along which little or no displacement of rock material has occurred.

Kaolinite: A common clay consisting mainly of hydrous aluminum silicate and closely related in chemical composition and crystal structure.

Kilowatt-hour (kWh): Use of electricity for one hour at a rate of 1000 watts.

Levelized Unit Cost: Capital and operating charges translated into an equivalent constant (or level) annual unit cost.

Light Water Reactor (LWR): May be either a BWR or PWR; uses as coolant ordinary water (H₂O) instead of heavy water (D₂O).

Lithification: The conversion of unconsolidated sediment into solid rock by processes such as compaction, cementation, and crystallization.

Lithology: The study of rocks. Also the character of a rock: its structure, color, mineral composition, grain size, and arrangement of its component parts.

Lithostatic pressure: The confining pressure at depth in the crust of the earth due to the weight of the overlying rocks.

Littoral: Belonging to, inhabiting or taking place on or near the shore of a body of water.

Low Enriched Uranium (LEU): Uranium containing less than 5% by weight but greater than 0.72% by weight ²³⁵U.

M&M Shaft: Men and Materials shaft at a mined repository.

Mafic: Pertaining to or composed dominantly of magnesium rock-forming silicates.

Magmatism: The development, movement, and solidification to igneous rock, of magma, a naturally occurring mobile rock material, generated within the earth and capable of intrusion and extrusion.

Maximum Individual, Maximum-Exposed Individual: A person whose location and habits tend to maximize his radiation dose.

Megawatts (MW): Millions of watts.

Mica: A group of silicate minerals of aluminum and other bases, especially potassium, magnesium, and iron, and characterized by great perfection of cleavage in one direction, that produces thin, tough, elastic plates.

Mixed-Oxide Fuel Fabrication Plant (MOX-FFP): Plant where uranium oxide and plutonium oxide are mixed and fabricated into fuel elements for use in nuclear power plants.

MOX: Mixed oxides (of uranium and plutonium).

MTHM: Metric tons of heavy metal (usually refers to reactor fuel, in which the heavy metals are uranium and plutonium).

Mucking and/or Settling Ponds: Ponds next to drilling operations where the excavated mud or slurry is placed; the sediment that settles at the bottom of these ponds is called muck.

Multibarrier: A system using the waste form, the container (canister), the overpack, the emplacement medium, and surrounding geologic media as multiple barriers to isolate the waste from the biosphere.

NAS: National Academy of Sciences.

NASA: National Aeronautics and Space Administration.

NCRP: National Council on Radiation Protection and Measurement.

- Neutron: Stable particle in a nucleus of very slightly greater mass than a proton but without nuclear change.
- NO_x: Oxides of nitrogen, specifically NO and NO₂.
- NRC: Nuclear Regulatory Commission.
- Nucleus: The inner core of the atom, consisting primarily of neutrons and protons, which make up almost the entire mass of the atom but only a minute part of its volume.
- Nuclide: A species of atom characterized by its mass number, atomic number, and nuclear energy state; to be regarded as a distinct nuclide the atom must be capable of existing for a measurable lifetime in its nuclear energy state.
- Olivine: An olive-green, common rock-forming ferromagnesian silicate mineral of mafic, Ultramafic, and low-silica igneous rocks.
- ONWI: Office of Nuclear Waste Isolation at Battelle Memorial Institute, Columbus, Ohio; under contract to DOE.
- Operations: Broad classification of waste management activities in terms of their basic function (e.g., waste storage, treatment, transportation or disposal).
- ORNL: Oak Ridge National Laboratory.
- Overpack: Secondary (or additional) external containment for packaged nuclear waste.
- Outcrop: A part of a body of rock that appears, bare and exposed, at the surface of the ground.
- Parent Nuclide: A radionuclide that upon disintegration yields a specified nuclide, either directly or as a later member of a radioactive decay series.
- Partition: To separate one (or more) element(s) from one (or more) other element(s). Examples include the separation of uranium and plutonium from each other, the separation of actinides and fission products in the waste, and the separation of one fission product from the other fission products.
- Perihelion: The point in the orbit of a celestial body that is closest to the sun.
- Permeability: The quality or state of being permeable. The relative ease with which a porous medium can transmit a liquid under a hydraulic gradient.
- Peridotite: A coarse-grained plutonic igneous rock composed chiefly of the mineral olivine but also containing considerable amounts of other ferromagnesian minerals.
- Plagioclase: The group of common rock-forming feldspar minerals; silicates of varying mixtures of sodium and calcium.
- Pluton: A body of intrusive igneous rock of any shape or size.
- Pluvial: Pertaining to a period of time in which rainfall or precipitation is abundant.
- PNL: Pacific Northwest Laboratory operated for DOE by Battelle Memorial Institute.
- Porosity: That property of a rock or soil which enables the rock or soil to contain water in voids or interstices, usually expressed in percentage or as a decimal fraction of void volume as compared to total volume.
- Pressurized Water Reactor (PWR): A reactor system that uses a pressurized water primary cooling system. Steam formed in a secondary cooling system is used to turn turbines to generate electricity.

Primary Wastes: Untreated initial wastes resulting from operation of fuel cycle facilities other than waste management facilities (wastes from operation of waste management facilities are secondary wastes).

Pyroxene: A group of dark rock-forming silicate minerals closely related in crystal form and analogous in chemical composition to the amphiboles; found chiefly in igneous rocks.

Rad: Radiation absorbed dose, the basic unit of absorbed dose of ionizing radiation. A dose of 1 rad is equivalent to the absorption of 100 ergs of radiation energy per gram of absorbing material.

Recharge: In hydrology, a source or means for replenishment of water withdrawn or discharged from an aquifer.

rem (roentgen equivalent man): A quantity used in radiation protection to express the effective dose equivalent for all forms of ionizing radiation. It is the product of the adsorbed dose in rads and factors related to relative biological effectiveness.

Remotely Handled Waste: Waste package having surface dose rate greater than 0.2 R/hr. Such packages require extensive shielding and/or remote handling to protect operating personnel. Remotely handled wastes were termed intermediate-level wastes in DOE/ET-0028 and DOE/ET-0029.

Repository (Federal): A Federally owned and operated facility for storage or disposal of specific types of waste from DOE sites and/or licensees.

Retrievability: Capability to remove waste from its place in isolation with approximately the same level of effort and radiation exposure as required to place the waste.

RH-TRU: Remotely handled TRU waste.

Risk (mathematical): Product of the consequences and the probability of the event's occurrence.

Roentgen: A unit for measuring gamma or "x-ray" radiation. The Roentgen is defined by measuring the effect of the radiation on air. It is that amount of gamma or x-rays required to produce ions carrying 1 electrostatic unit of charge in 0.001293 g of dry air under standard conditions; $1 R = 2.58 \times 10^{-4}$ coulomb/kg.

RWSF: Retrievable waste storage facility.

Scrubbers: An apparatus that chemically removes impurities from exhaust gas emissions.

Secondary Wastes: Wastes that result from applying waste treatment technologies to primary wastes.

Sedimentary Basin: A geologically depressed area that has thick sediments in the interior and thinner sediments at the edges.

Seismicity: The phenomenon of earth movements as manifested by earthquakes.

SFPF: Spent fuel packaging facility.

Shield: A continental segment of the earth's crust which has been relatively stable over a long period of time and which has exposed crystalline rocks mostly of Precambrian age; in general, representing the oldest rocks of the continent.

Shielding: A material interposed between a source of radiation and personnel for protection against the danger of radiation. Commonly used shielding materials are concrete, water and lead.

Shipping Cask: A specially designed container used for shipping radioactive materials.

SHLW: Solidified high-level waste.

- Short-Lived Nuclides:** Radioactive isotopes with relatively short half-lives. Usage for some isotopes varies with the concept being considered (e.g., isotopes with 5-50 year half-lives are short lived in the context of geologic disposal but long lived in the context of predisposal operations).
- Slurry:** A fluid mixture or suspension of insoluble material.
- Solidification:** Conversion of liquid radioactive waste to a dry, stable solid.
- Source Terms:** The quantity of radioactive material (or other pollutant) released to the environment at its point of release (source).
- Spent Fuel (SF):** Nuclear reactor fuel that has been used to the extent that it can no longer be used efficiently in a nuclear power plant.
- Stock:** An igneous intrusion less than 100 km² (40 mi²) in surface exposure.
- Storage:** Retention of waste in some type of manmade device in a manner permitting retrieval.
- Strain:** Deformation resulting from applied stress; proportional to stress.
- Stratum:** Sedimentary bed or layer, regardless of thickness, of homogeneous or gradational lithology.
- Syncline:** A fold, the core of which contains stratigraphically younger rocks, and which, in simplest form, is elongate and concave upward with the two limbs dipping toward each other.
- Tailings:** The part of any ore that is regarded as too poor to be treated further.
- Tails:** In the case of uranium it refers to the depleted uranium left after enrichment operations.
- TBP:** Tributyl phosphate, a solvent used in the PUREX fuel reprocessing process.
- Technologies:** Specific methods for implementing concepts. An example is calcination of liquid high-level waste by using a spray calciner.
- Tectonic:** Of, pertaining to, or designating the processes causing, and the rock structures resulting from, deformation of the earth's crust.
- Tectonism (diastrophism):** Crustal movement produced by earth forces, such as the formation of plateaus and mountain ranges; the structural behavior of an element of the earth's crust during, or between, major cycles of sedimentation.
- Theoretical Density (TD):** Maximum density attainable for any given material.
- Thermal Regime:** The area adjacent to a heat source which is affected by that source.
- Trajectory:** The curve that an object describes in space in traveling from one point to another.
- Transmissivity:** Volume of water flowing through a 1-ft width of aquifer of given thickness under a unit gradient (1 ft vertically for each 1 ft laterally) and at the viscosity prevailing in the field. Mathematically, it is the product of permeability and aquifer thickness.
- Transmutation:** A nuclear process in which one nuclide is transformed into the nuclide of a different element. This can be accomplished by bombardment with neutrons or other nuclear particles.

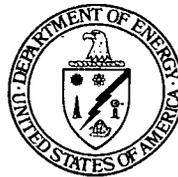
- Transportation:** Movement of materials between sites. Intra-site movement is not considered. Includes alternative methods for packaging, handling, and transport of waste materials and plutonium compounds. Concepts include all conventional methods of land and water transport required by the waste management system.
- Transuranic (TRU) elements:** Elements with atomic number greater than 92. They include, among others, neptunium, plutonium, americium, and curium.
- Transuranic Waste:** Waste material measured or assumed to contain more than a specified concentration of transuranic elements. For purposes of this Statement, TRU waste is waste from locations that might cause contamination levels above 10 nanocuries of transuranic alpha activity per gram of waste.
- Treatment:** Operations intended to benefit safety or economy by changing the waste characteristics.
- Ultramafic:** Pertaining to igneous rocks composed chiefly of ferromagnesian dark minerals.
- Uplift:** A structurally high area in the crust, produced by movements that raise or upthrust the rocks, as in a dome or arch.
- Vital Areas:** The code of Federal Regulations (10 CFR 73), defines equipment items, systems, devices, and materials whose failure, destruction or release could directly endanger the public health and safety by exposure to radiation defined as "vital". Areas containing such items or materials (e.g., spent fuel or high-level waste) are defined as "vital" areas and subject to special protection measures.
- Waste Immobilization:** Process of converting waste to a stable, solid and relatively insoluble form.
- Waste Isolation Pilot Plant (WIPP):** A Defense repository proposed for a site in South-eastern New Mexico.
- Waste Management:** The planning, execution and surveillance of essential functions related to the control of radioactive (and nonradioactive) waste, including treatment, transportation, storage, surveillance, and isolation.
- Water Table:** The upper surface of the zone of water saturation in the subsurface, at which the pressure is equal to atmospheric pressure; the upper surface of an unconfined aquifer.

FINAL

ENVIRONMENTAL IMPACT STATEMENT

**Management of
Commercially Generated
Radioactive Waste**

**Volume 2
Appendices**



October 1980

**U.S. Department of Energy
Assistant Secretary for Nuclear Energy
Office of Nuclear Waste Management
Washington, D.C. 20545**

VOLUME 2CONTENTS

APPENDIX A - WASTE-MANAGEMENT SYSTEMS SUPPLEMENTARY DATA	A.1
A.1 WASTE LOGISTICS TABLES	A.1
A.2 RADIOACTIVE INVENTORY TABLES	A.30
A.3 HEAT GENERATION RATE TABLES	A.49
A.4 HAZARD INDEX TABLES	A.68
A.5 SUPPLEMENTARY DOSE TABLES	A.87
A.6 RESOURCE COMMITMENTS	A.92
A.7 TRANSPORTATION REQUIREMENTS	A.96
A.8 SUPPLEMENTARY PREDISPOSAL COST DATA	A.99
A.9 SUPPLEMENTARY SYSTEM COST DATA	A.104
A.10 SYSTEM REPOSITORY REQUIREMENTS	A.119
APPENDIX B - GEOLOGIC DISPOSAL SUPPLEMENTARY INFORMATION	B.1
B.1 DEPTH OF REPOSITORY	B.1
B.2 DIMENSIONS AND PROPERTIES OF HOST ROCKS AND MEDIA	B.4
B.3 SEISMIC, TECTONIC AND MAGNETIC CONSIDERATIONS	B.8
B.4 HYDROLOGIC CONSIDERATIONS	B.11
B.5 NATURAL RESOURCE CONSIDERATIONS	B.14
B.6 MULTIPLE GEOLOGIC BARRIERS	B.15
B.7 THE SITE SELECTION PROCESS	B.21
REFERENCES FOR APPENDIX B	B.26
APPENDIX C - RADIOLOGICAL STANDARDS	C.1
C.1 "AS LOW AS REASONABLY ACHIEVABLE" APPLICATION	C.3
C.2 DERIVED LIMITS AND ACTION LEVELS	C.4
REFERENCES FOR APPENDIX C	C.7
APPENDIX D - MODELS USED IN DOSE CALCULATIONS	D.1
D.1 DOSE TO REGIONAL POPULATION	D.1
D.2 DOSE TO WORLDWIDE POPULATION	D.7
REFERENCES FOR APPENDIX D	D.14

CONTENTS (contd)

APPENDIX E - RADIOLOGICALLY RELATED HEALTH EFFECTS	E.1
E.1 LATE SOMATIC EFFECTS	E.2
E.2 GENETIC EFFECTS	E.7
E.3 SUMMARY	E.10
E.4 SPECIFIC CONSIDERATION OF HEALTH EFFECTS FROM TRANSURANICS	E.12
E.5 SPECIFIC CONSIDERATION OF HEALTH EFFECTS FROM KRYPTON-85	E.14
E.6 SPECIFIC CONSIDERATION OF HEALTH EFFECTS FROM TRITIUM	E.15
E.7 SPECIFIC CONSIDERATION OF HEALTH EFFECTS FROM CARBON-14	E.18
REFERENCES FOR APPENDIX E	E.17
APPENDIX F - REFERENCE ENVIRONMENT FOR ASSESSING ENVIRONMENTAL IMPACTS	F.1
F.1 LOCATION OF SITE	F.1
F.2 REGIONAL DEMOGRAPHY AND LAND USE	F.2
F.3 GEOLOGY	F.4
F.4 HYDROLOGY	F.5
F.5 METEOROLOGY	F.8
F.6 PATHWAY PARAMETERS RELEVANT TO RADIOLOGICAL DOSE CALCULATIONS	F.10
APPENDIX G - REFERENCE SITES FOR ASSESSING SOCIAL AND ECONOMIC IMPACTS	G.1
G.1 CRITERIA FOR REFERENCE SITE SELECTION	G.1
G.2 CHARACTERISTICS OF REFERENCE SITES	G.2
APPENDIX H - HAZARD INDICES	H.1
REFERENCES FOR APPENDIX H	H.4
APPENDIX I - COMPARISON OF DEFENSE PROGRAM WASTE TO COMMERCIAL RADIOACTIVE WASTE	I.1
I.1 HIGH-LEVEL WASTE COMPARISON	I.1
I.2 TRU WASTE COMPARISONS	I.2
APPENDIX J - NOT USED ^(a)	
APPENDIX K - GEOLOGIC REPOSITORY DESIGN CONSIDERATIONS	K.1
K.1 THERMAL CRITERIA	K.1

(a) Essential information from this appendix appears in Volume 1 of the final Statement.

CONTENTS (contd)

K.2	REMOVAL OF EMPLACED WASTE	K.23
K.3	ENGINEERED SORPTION BARRIERS	K.26
	REFERENCES FOR APPENDIX K	K.30
APPENDIX L	- SUPPORTING RESEARCH AND DEVELOPMENT	L.1
L.1	GEOLOGIC SITE SELECTION	L.1
L.2	HOST ROCK PROPERTIES	L.4
L.3	THERMAL AND RADIATION EFFECTS	L.6
L.4	REPOSITORY PERFORMANCE	L.7
	REFERENCES FOR APPENDIX L	L.9
APPENDIX M	- BIBLIOGRAPHY FOR ALTERNATIVE CONCEPTS, SECTION 6.1	M.1
APPENDIX N	- WASTES FROM THORIUM-BASED FUEL CYCLE ALTERNATIVES	N.1
	REFERENCES FOR APPENDIX N	N.4
APPENDIX O	- NOT USED(a)		
APPENDIX P	- STABILITY OF MINERALS THAT COULD CONTAIN RADIONUCLIDES	P.1
P.1	PHYSICO-CHEMICAL PRINCIPLES	P.1
P.2	DISCUSSION OF MINERAL GROUPS	P.14
P.3	MINERAL TABLES	P.35
P.4	METAMICTIZATION	P.44
	REFERENCES FOR APPENDIX P	P.49

(a) Essential information from this appendix appears in Volume 1 of the final Statement.

VOLUME 2FIGURES

B.6.1	Bedded Salt Deposits and Salt Domes in the United States	B.16
B.6.2	Granitic Rock in the United States	B.18
B.6.3	Representative Shale Units in the United States	B.19
B.6.4	Potential Repository Basalts in the United States	B.20
B.7.1	Site Selection Process, Stage I	B.22
B.7.2	Site Selection Process, Stage II	B.23
B.7.3	Site-Selection Process, Stage III	B.24
B.7.4	Additional Data Base Requirements	B.25
C.2.1	Relationship of Operating Levels, Action Levels, and Concentration Guides	C.5
F.4.1	Daily Average and Extreme River Flows at the Reference Site	F.5
F.4.2	Daily Average and Extreme Water Temperatures at the Reference Site	F.5
F.4.3	River Flow Duration Data for R River at the Reference Site	F.6
F.6.1	Pathways for Radiation Exposure of Man	F.10
K.1.1	Formation Temperature versus Depth and Time for Repository in Salt--Once-Through Fuel Cycle	K.10
K.1.2	Formation Temperature versus Depth and Time for Repository in Salt--Reprocessing Fuel Cycle	K.10
K.1.3	Formation Temperature versus Depth and Time for Repository in Granite--Once-Through Fuel Cycle	K.11
K.1.4	Formation Temperature versus Depth and Time for Repository in Granite--Reprocessing Fuel Cycle	K.11
K.1.5	Formation Temperature versus Depth and Time for Repository in Shale--Once-Through Fuel Cycle	K.12
K.1.6	Formation Temperature versus Depth and Time for Repository in Shale--Reprocessing Fuel Cycle	K.12
K.1.7	Formation Temperature versus Depth and Time for Repository in Basalt--Once-Through Fuel Cycle	K.13
K.1.8	Formation Temperature versus Depth and Time for Repository in Basalt--Reprocessing Fuel Cycle	K.13
K.1.9	Very-Near-Field Temperatures versus Time for Repository in Salt--Once-Through Fuel Cycle	K.14
K.1.10	Very-Near-Field Temperatures versus Time for Repository in Salt--Reprocessing Fuel Cycle	K.14

FIGURES (contd)

K.1.11	Very-Near-Field Temperatures versus Time for Repository in Granite--Once-Through Fuel Cycle	K.15
K.1.12	Very-Near-Field Temperatures versus Time for Repository in Granite--Reprocessing Fuel Cycle	K.15
K.1.13	Very-Near-Field Temperatures versus Time for Repository in Shale--Once-Through Fuel Cycle	K.16
K.1.14	Very-Near-Field Temperatures versus Time for Repository in Shale--Reprocessing Fuel Cycle	K.16
K.1.15	Very-Near-Field Temperatures versus Time for Repository in Basalt--Once-Through Fuel Cycle	K.17
K.1.16	Very-Near-Field Temperatures versus Time for Repository in Basalt--Reprocessing Fuel Cycle	K.17

VOLUME 2TABLES

A.1.1	Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 1, MTU No Repository	A.2
A.1.2	Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 1, MTU 1990 Repository	A.3
A.1.3	Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 1, MTU 2010 Repository	A.4
A.1.4	Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 1, MTU 2030 Repository	A.5
A.1.5	Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 2, MTU No Repository	A.6
A.1.6	Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 2, MTU 1990 Repository	A.7
A.1.7	Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 2, MTU 2010 Repository	A.8
A.1.8	Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 2, MTU 2030 Repository	A.9
A.1.9	Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 3, MTU No Repository	A.10
A.1.10	Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 3, MTU 1990 Repository	A.11
A.1.11	Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 3, MTU 2010 Repository	A.12
A.1.12	Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 3, MTU 2030 Repository	A.13
A.1.13	Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 4, MTU 2000 Repository	A.14
A.1.14	Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 4, MTU 2020 Repository	A.15
A.1.15	Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 5, MTU 2000 Repository	A.16
A.1.16	Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 5, MTU 2020 Repository	A.17
A.1.17	Spent Fuel Logistics for the Reprocessing Fuel Cycle--Growth Case 3, MTU 1990 Reprocessing	A.19
A.1.18	Spent Fuel Logistics for the Reprocessing Fuel Cycle--Growth Case 3, MTU 2010 Reprocessing	A.21
A.1.19	Spent Fuel Logistics for the Reprocessing Fuel Cycle--Growth Case 4, MTU 2000 Reprocessing	A.23

TABLES (contd)

A.1.20	Spent Fuel Logistics for the Reprocessing Fuel Cycle--Growth Case 5, MTU 2000 Reprocessing	A.25
A.1.21	Number of Containers Sent to Repository in Once-Through Cases	A.27
A.1.22	Number of Containers Sent to Repository in Reprocessing Cases	A.28
A.1.23	Age of HLW at Time of Disposal	A.29
A.2.1a	Radioactivity Inventory--Once-Through Cycle--Growth Case 1, Curies Fission and Activation Products	A.31
A.2.1b	Radioactivity Inventory--Once-Through Cycle--Growth Case 1, Curies Actinides	A.32
A.2.2a	Radioactivity Inventory--Once-Through Cycle--Growth Case 2, Curies Fission and Activation Products	A.33
A.2.2b	Radioactivity Inventory--Once-Through Cycle--Growth Case 2, Curies Actinides	A.34
A.2.3a	Radioactivity Inventory--Once-Through Cycle--Growth Case 3, Curies Fission and Activation Products	A.35
A.2.3b	Radioactivity Inventory--Once-Through Cycle--Growth Case 3, Curies Actinides	A.36
A.2.4a	Radioactivity Inventory--Once-Through Cycle--Growth Case 4, Curies Fission and Activation Products	A.37
A.2.4b	Radioactivity Inventory--Once-Through Cycle--Growth Case 4, Curies Actinides	A.38
A.2.5a	Radioactivity Inventory--Once-Through Cycle--Growth Case 5, Curies Fission and Activation Products	A.39
A.2.5b	Radioactivity Inventory--Once-Through Cycle--Growth Case 5, Curies Actinides	A.40
A.2.6a	Radioactivity Inventory--Reprocessing Cycle--Growth Case 3 - 1990 Reprocessing Startup, Curies Fission and Activation Products	A.41
A.2.6b	Radioactivity Inventory--Reprocessing Cycle--Growth Case 3 - 1990 Reprocessing Startup, Curies Actinides	A.42
A.2.7a	Radioactivity Inventory--Reprocessing Cycle--Growth Case 3 - 2010 Reprocessing Startup, Curies Fission and Activation Products	A.43
A.2.7b	Radioactivity Inventory--Reprocessing Cycle--Growth Case 3 - 2010 Reprocessing Startup, Curies Actinides	A.44
A.2.8a	Radioactivity Inventory--Reprocessing Cycle--Growth Case 4 - 2000 Reprocessing Startup, Curies Fission and Activation Products	A.45
A.2.8b	Radioactivity Inventory--Reprocessing Cycle--Growth Case 4 - 2000 Reprocessing Startup, Curies Actinides	A.46

TABLES (contd)

A.2.9a	Radioactivity Inventory--Reprocessing Cycle--Growth Case 5 - 2000 Reprocessing Startup, Curies Fission and Activation Products	A.47
A.2.9b	Radioactivity Inventory--Reprocessing Cycle--Growth Case 5 - 2000 Reprocessing Startup, Curies Actinides	A.48
A.3.1a	Heat Generation Rates--Once-Through Cycle--Growth Case 1, Watts Fission and Activation Products	A.50
A.3.1b	Heat Generation Rates--Once-Through Cycle, Growth Case 1, Watts Actinides	A.51
A.3.2a	Heat Generation Rates--Once-Through Cycle--Growth Case 2, Watts Fission and Activation Products	A.52
A.3.2b	Heat Generation Rates--Once-Through Cycle--Growth Case 2, Watts Actinides	A.53
A.3.3a	Heat Generation Rates--Once-Through Cycle--Growth Case 3, Watts Fission and Activation Products	A.54
A.3.3b	Heat Generation Rates--Once-Through Cycle--Growth Case 3, Watts Actinides	A.55
A.3.4a	Heat Generation Rates--Once-Through Cycle--Growth Case 4, Watts Fission and Activation Products	A.56
A.3.4b	Heat Generation Rates--Once Through Cycle--Growth Case 4, Watts Actinides	A.57
A.3.5a	Heat Generation Rates--Once-Through Cycle--Growth Case 5, Watts Fission and Activation Products	A.58
A.3.5b	Heat Generation Rates--Once Through Cycle--Growth Case 5, Watts Actinides	A.59
A.3.6a	Heat Generation Rates--Reprocessing Cycle--Growth Case 3-- 1990 Reprocessing Startup, Watts Fission and Activation Products	A.60
A.3.6b	Heat Generation Rates, Reprocessing Cycle--Growth Case 3, 1990 Reprocessing Startup, Watts Actinides	A.61
A.3.7a	Heat Generation Rates, Reprocessing Cycle, Growth Case 3, 2010 Reprocessing Startup, Watts Fission and Activation Products	A.62
A.3.7b	Heat Generation Rates--Reprocessing Cycle--Growth Case 3 - 2010 Reprocessing Startup, Watts Actinides	A.63
A.3.8a	Heat Generation Rates--Reprocessing Cycle--Growth Case 4 - 2000 Reprocessing Startup, Watts Fission and Activation Products	A.64
A.3.8b	Heat Generation Rates--Reprocessing Cycle--Growth Case 4 - 2000 Reprocessing Startup, Watts Actinides	A.65
A.3.9a	Heat Generation Rates--Reprocessing Cycle--Growth Case 5 - 2000 Reprocessing Startup, Watts Fission and Activation Products	A.66

TABLES (contd)

A.3.9b	Heat Generation Rates--Reprocessing Cycle--Growth Case 5 - 2000 Reprocessing Startup, Watts Actinides	A.67
A.4.1a	Hazard Index--Once-Through Cycle--Growth Case 1, m ³ water/MTHM Fission and Activation Products	A.69
A.4.1b	Hazard Index--Once-Through Cycle--Growth Case 1, m ³ water/MTHM Actinides	A.70
A.4.2a	Hazard Index--Once-Through Cycle--Growth Case 2, m ³ water/MTHM Fission and Activation Products	A.71
A.4.2b	Hazard Index--Once-Through Cycle--Growth Case 2, m ³ water/MTHM Actinides	A.72
A.4.3a	Hazard Index--Once-Through Cycle--Growth Case 3, m ³ water/MTHM Fission and Activation Products	A.73
A.4.3b	Hazard Index--Once-Through Cycle--Growth Case 3, m ³ water/MTHM Actinides	A.74
A.4.4a	Hazard Index--Once-Through Cycle--Growth Case 4, m ³ water/MTHM Fission and Activation Products	A.75
A.4.4b	Hazard Index--Once-Through Cycle--Growth Case 4, m ³ water/MTHM Actinides	A.76
A.4.5a	Hazard Index--Once-Through Cycle--Growth Case 5, m ³ water/MTHM Fission and Activation Products	A.77
A.4.5b	Hazard Index--Once-Through Cycle--Growth Case 5, m ³ water/MTHM Actinides	A.78
A.4.6a	Hazard Index--Reprocessing Cycle--Growth Case 3--1990 Reprocessing Startup, m ³ water/MTHM Fission and Activation Products	A.79
A.4.6b	Hazard Index--Reprocessing Cycle--Growth Case 3--1990 Reprocessing Startup, m ³ water/MTHM Actinides	A.80
A.4.7a	Hazard Index--Reprocessing Cycle--Growth Case 3--2010 Reprocessing Startup, m ³ water/MTHM Fission and Activation Products	A.81
A.4.7b	Hazard Index--Reprocessing Cycle--Growth Case 3--2010 Reprocessing Startup, m ³ water/MTHM Actinides	A.82
A.4.8a	Hazard Index--Reprocessing Cycle--Growth Case 4--2000 Reprocessing Startup, m ³ water/MTHM Fission and Activation Products	A.83
A.4.8b	Hazard Index--Reprocessing Cycle--Growth Case 4--2000 Reprocessing Startup, m ³ water/MTHM Actinides	A.84
A.4.9a	Hazard Index--Reprocessing Cycle--Growth Case 5--2000 Reprocessing Startup, m ³ water/MTHM Fission and Activation Products	A.85
A.4.9b	Hazard Index--Reprocessing Cycle--Growth Case 5--2000 Reprocessing Startup, m ³ water/MTHM Actinides	A.86
A.5.1a	Whole-Body Dose to the Population for the Once-Through Cycle, Man-Rem	A.88

TABLES (contd)

A.5.1b	Bone Dose to the Population for the Once-Through Cycle, Man-Rem	A.88
A.5.1c	Lung Dose to the Population for the Once-Through Cycle, Man-Rem	A.89
A.5.1d	Thyroid Dose to the Population for the Once-Through Cycle, Man-Rem	A.89
A.5.2a	Whole-Body Dose to the Population for the Reprocessing Cycle, Man-Rem	A.90
A.5.2b	Bone Dose to the Population for the Reprocessing Cycle, Man-Rem	A.90
A.5.2c	Lung Dose to the Population for the Reprocessing Cycle, Man-Rem	A.91
A.5.2d	Thyroid Dose to the Population for the Reprocessing Cycle, Man-Rem	A.91
A.6.1	Resource Commitments With the Once-Through Cycle	A.93
A.6.2	Resource Commitments with the Reprocessing Cycle	A.94
A.6.3	Resource Commitments for Shipping Casks	A.95
A.7.1	Transportation Requirements Using the Once-Through Fuel Cycle	A.97
A.7.2	Transportation Requirements Using the Reprocessing Cycle	A.98
A.8.1	Cost Estimates for Treatment and Storage of Spent Fuel	A.100
A.8.2	Cost Estimates for Treatment of Waste from Uranium and Plutonium Recycle	A.101
A.8.3	Cost Estimates for Interim Storage of Waste from Uranium and Plutonium Recycle	A.102
A.8.4	Cost Estimates for Waste Transportation	A.103
A.9.1a	Allocation of Total-System Waste Management Unit Costs with the Once-Through Cycle Using a 0% Discount Rate, mills/kWh	A.105
A.9.1b	Allocation of Total-System Waste Management Unit Costs with the Once-Through Cycle Using a 7% Discount Rate, mills/kWh	A.106
A.9.1c	Allocation of Total-System Waste Management Unit Costs with the Once-Through Cycle Using a 10% Discount Rate, mills/kWh	A.107
A.9.2a	Allocation of Total-System Waste Management Unit Costs with the Reprocessing Cycle Using a 0% Discount Rate, mills/kWh	A.108
A.9.2b	Allocation of Total-System Waste Management Unit Costs with the Reprocessing Cycle Using a 7% Discount Rate, mills/kWh	A.109
A.9.2c	Allocation of Total-System Waste Management Unit Costs with the Reprocessing Cycle Using a 10% Discount Rate, mills/kWh	A.110
A.9.3a	Repository Media Effect on Total-System Waste Management Unit Cost with the Once-Through cycle Using a 0% Discount Rate	A.111
A.9.3b	Repository Media Effect on Total-System Waste Management Unit Cost with the Once-Through cycle Using a 7% Discount Rate	A.112

TABLES (contd)

A.9.3c	Repository Media Effect on Total-System Waste Management Unit Cost with the Once-Through cycle Using a 10% Discount Rate	A.113
A.9.4a	Repository Media Effect on Total-System Waste Management Unit Costs with the Reprocessing cycle Using a 0% Discount Rate	A.114
A.9.4b	Repository Media Effect on Total-System Waste Management Unit Costs with the Reprocessing cycle Using a 7% Discount Rate	A.115
A.9.4c	Repository Media Effect on Total-System Waste Management Unit Costs with the Reprocessing cycle Using a 10% Discount Rate	A.116
A.9.5	Estimated Research and Development Costs for Predisposal Management for a 1990 Repository Start, \$ Millions	A.117
A.9.6	Estimated Research and Development Cost (including site verification) for Waste Isolation	A.118
A.10.1	Repository Requirements for Once-Through Cycle	A.120
A.10.2	Repository Requirements for Recycle Cases	A.121
B.2.1	Physical Properties of Media	B.6
B.6.1	Average Chemical Composition by Oxides for Representative Disposal Media	B.19
C.2.1	Comparison Chart of Radiation Standards and Recommendations	C.4
D.2.1	Total-Body Dose Factors, and Dose Commitment Factors for the World Population	D.8
D.2.2	70-Year World Population Dose Commitment from a 1-Year Chronic Release, man-rem/70 Years per Ci/yr Released	D.13
E.1.1	Comparison of Various Estimates of Cancer Deaths per Million Man-Rem	E.4
E.1.2	Health Effects Risk Factors Employed in this Statement	E.6
E.2.1	Estimates of Genetic Effects of Radiation Over All Generations	E.8
E.4.1	Comparison of Transuranic Health Risk Estimates	E.13
F.2.1	Projected Year 2000 Population in Reference Environment	F.2
F.4.1	R River Water Chemistry Summary of 12 Monthly Samples	F.7
F.5.1	Monthly Temperature Statistics	F.8
F.5.2	Mean Monthly Relative Humidity Percent	F.8
F.5.3	Annual Average Joint Frequency Distribution, Percent of Occurrence	F.9
G.2.1	Selected Data Characteristics of Three Reference Sites, Socioeconomic Impact Analysis	G.3
H.1	Hazard Indices	H.2
K.1.1	Thermal and Thermomechanical Limits for Conceptual Design Studies	K.2

TABLES (contd)

K.1.2	Thermal Load Limits for Conceptual Repository Designs	K.7
K.1.3	Cummulative Heat Generated by 10-Yr-Old Spent Fuel and High-Level Waste	K.7
K.1.4	Thermal Loadings Achieved at Conceptual Repositories	K.9
K.1.5	Material Properties	K.18
K.1.6	Thermal Conductivities	K.18
K.1.7	Thermal Loading Limits for Waste Repositories	K.19
K.1.8	Thermal Loadings Used	K.19
K.1.9	Repository Capacities as a Function of Waste Age	K.20
K.1.10	Maximum Near-Field Temperatures with Spent Fuel	K.20
K.1.11	Maximum Near-Field Temperatures with HLW	K.21
K.1.12	Maximum Far-Field Temperature Increases	K.21
K.1.13	Heat Generation Rates for Spent Fuel and High-level Wastes	K.22
K.2.1	Near-Field Local Thermal Densities for 25-Year Ready Retrievability of 10-year-old Spent Fuel	K.24
K.3.1	Predominant Solution Species of Elements Without Organic Ligands	K.27
K.3.2	Factors Reported to Effect Adsorption of Radioelements Over the pH Range of 4 to 9	K.28
K.3.3	Thermal Stabilities and Cation Exchange Capacities of Several Clay Minerals and Zeolites	K.28
K.3.4	Retention Time Ranges on 1-m Barriers for Several Radionuclides	K.29
K.3.5	Barrier Depth (m) Required to Retard Various Radionuclides 30 Half-Lives	K.29
P.1.1	Mobilities of the Common Cations	P.3
P.1.2	Weathering Sequence of Clay-Size Minerals in Soils and Sedimentary Deposits	P.4
P.1.3	Persistence Order of Minerals	P.4
P.1.4	Placer Minerals	P.7
P.1.5	Detrital Minerals	P.8
P.1.6	Selected Ionic Radii	P.10
P.2.1	Borosilicate and Berylosilicate Minerals	P.20
P.2.2	Zirconosilicate and Titanosilicate Minerals	P.20
P.2.3	Rare-Earth Silicate Minerals	P.21

TABLES (contd)

P.2.4	Uranyl Phosphates, Arsenates, Vanadates	P.32
P.2.5	The Autunite Family	P.33
P.3.1	Selected Host Minerals for Radionuclides	P.36
P.3.2	Systematic Tabulations of Metamict Minerals	P.42
P.4.1	Uranium and Thorium Content (wt%) of Non-Metamict and Metamict AB_2O_6 -- Type Nb-Ta-Ti Oxides	P.46
P.4.2	Radioactive Minerals Reported as Always Crystalline	P.46

APPENDIX A

WASTE-MANAGEMENT SYSTEMS SUPPLEMENTARY DATA

Appendix A contains supplementary data on the waste management systems simulation and related information. The data are presented in tables; types of data included are:

	<u>Tables</u>
Waste Logistics Tables	A.1.1 - A.1.23
Radioactive Inventory Tables	A.2.1a - A.2.9b
Heat Generation Rate Tables	A.3.1a - A.3.9b
Hazard Index Tables	A.4.1a - A.4.9b
Supplementary Dose Tables	A.5.1a - A.5.2d
Resource Commitments	A.6.1 - A.6.3
Transportation Requirements	A.7.1 - A.7.2
Supplementary Predisposal Cost Data	A.8.1 - A.8.4
Supplementary System Cost Data	A.9.1a - A.9.6
System Repository Requirements	A.10.1 - A.10.2

Brief descriptions of the types of data are given at the beginning of each section. The associated tables then follow.

A.1 WASTE LOGISTICS TABLES

The spent fuel logistics tables (A.1.1 through A.1.23) show the disposition and transportation of spent fuel in metric tons of heavy metal (MTHM) as a function of time. These tables correspond with the graphs of repository inventories shown in Chapter 7. A table is provided for each of the cases analyzed in both the once-through and the reprocessing cycles. Total waste quantities for disposal in the reprocessing cases are shown in Tables A.1.21 and A.1.22. The age of the HLW at the time of disposal is shown in Table A.1.23.

TABLE A.1.1. Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 1, MTU

No Repository

YEAR	REACTOR DISCHARGE	REACTOR STORAGE	SHIPMENT REACTOR TO APR	SHIPMENT REACTOR TO REPOSITORY	APR INVENTORY	SHIPMENT APR TO REPOSITORY	REPOSITORY			YEAR
							RECEIPTS	INVENTORY	RECEIVING AGE, YEARS	
1980	1160.	7196.	0.	0.	0.	0.	0.	0.	0.0	1980
1981	2760.	9956.	0.	0.	0.	0.	0.	0.	0.0	1981
1982	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1982
1983	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1983
1984	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1984
1985	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1985
1986	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1986
1987	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1987
1988	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1988
1989	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1989
1990	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1990
1991	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1991
1992	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1992
1993	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1993
1994	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1994
1995	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1995
1996	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1996
1997	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1997
1998	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1998
1999	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1999
2000	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2000
2001	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2001
2002	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2002
2003	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2003
2004	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2004
2005	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2005
2006	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2006
2007	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2007
2008	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2008
2009	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2009
2010	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2010
2011	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2011
2012	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2012
2013	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2013
2014	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2014
2015	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2015
2016	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2016
2017	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2017
2018	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2018
2019	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2019
2020	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2020
2021	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2021
2022	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2022
2023	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2023
2024	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2024
2025	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2025
2026	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2026
2027	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2027
2028	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2028
2029	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2029
2030	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2030
2031	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2031
2032	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2032
2033	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2033
2034	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2034
2035	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2035
2036	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2036
2037	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2037
2038	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2038
2039	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2039
2040	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2040
RAIL SHIPMENTS *			0.0	0.0		0.0				
TRUCK SHIPMENTS *			0.0	0.0						

TABLE A.1.2. Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 1, MTU

1990 Repository

YEAR	REACTOR DISCHARGE	REACTOR STORAGE	SHIPMENT REACTOR TO AFR	SHIPMENT REACTOR TO REPOSITORY	AFR INVENTORY	SHIPMENT AFR TO REPOSITORY	REPOSITORY			YEAR
							RECEIPTS	INVENTORY	RECEIVING AGE, YEARS	
1980	1160.	7196.	0.	0.	0.	0.	0.	0.	0.0	1980
1981	2760.	9956.	0.	0.	0.	0.	0.	0.	0.0	1981
1982	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1982
1983	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1983
1984	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1984
1985	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1985
1986	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1986
1987	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1987
1988	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1988
1989	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1989
1990	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1990
1991	0.	9256.	0.	700.	0.	0.	700.	700.	18.0	1991
1992	0.	7956.	0.	1300.	0.	0.	1300.	2000.	16.4	1992
1993	0.	5956.	0.	2000.	0.	0.	2000.	4000.	15.3	1993
1994	0.	3956.	0.	2000.	0.	0.	2000.	6000.	14.4	1994
1995	0.	2760.	0.	1196.	0.	0.	1196.	7196.	14.0	1995
1996	0.	0.	0.	2760.	0.	0.	2760.	9956.	14.0	1996
1997	0.	0.	0.	0.	0.	0.	0.	9956.	0.0	1997
1998	0.	0.	0.	0.	0.	0.	0.	9956.	0.0	1998
1999	0.	0.	0.	0.	0.	0.	0.	9956.	0.0	1999
2000	0.	0.	0.	0.	0.	0.	0.	9956.	0.0	2000
2001	0.	0.	0.	0.	0.	0.	0.	9956.	0.0	2001
RAIL SHIPMENTS ■			0.0	2274.2		0.0				
TRUCK SHIPMENTS ■			0.0	2315.3						

TABLE A.1.3. Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 1, MTU

2010 Repository

YEAR	REACTOR DISCHARGE	REACTOR STORAGE	SHIPMENT REACTOR TO AFR	SHIPMENT REACTOR TO REPOSITORY	AFR INVENTORY	SHIPMENT AFR TO REPOSITORY	REPOSITORY			YEAR
							RECEIPTS	INVENTORY	RECEIVING AGE, YEARS	
1980	1160.	7196.	0.	0.	0.	0.	0.	0.	0.0	1980
1981	2760.	9956.	0.	0.	0.	0.	0.	0.	0.0	1981
1982	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1982
1983	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1983
1984	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1984
1985	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1985
1986	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1986
1987	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1987
1988	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1988
1989	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1989
1990	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1990
1991	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1991
1992	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1992
1993	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1993
1994	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1994
1995	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1995
1996	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1996
1997	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1997
1998	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1998
1999	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	1999
2000	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2000
2001	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2001
2002	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2002
2003	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2003
2004	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2004
2005	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2005
2006	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2006
2007	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2007
2008	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2008
2009	0.	9956.	0.	0.	0.	0.	0.	0.	0.0	2009
2010	0.	9256.	0.	700.	0.	0.	700.	700.	38.0	2010
2011	0.	7956.	0.	1300.	0.	0.	1300.	2000.	36.4	2011
2012	0.	5956.	0.	2000.	0.	0.	2000.	4000.	35.3	2012
2013	0.	3956.	0.	2000.	0.	0.	2000.	6000.	34.4	2013
2014	0.	2760.	0.	1196.	0.	0.	1196.	7196.	34.0	2014
2015	0.	0.	0.	2760.	0.	0.	2760.	9956.	34.0	2015
2016	0.	0.	0.	0.	0.	0.	0.	9956.	0.0	2016
2017	0.	0.	0.	0.	0.	0.	0.	9956.	0.0	2017
2018	0.	0.	0.	0.	0.	0.	0.	9956.	0.0	2018
2019	0.	0.	0.	0.	0.	0.	0.	9956.	0.0	2019
2020	0.	0.	0.	0.	0.	0.	0.	9956.	0.0	2020
2021	0.	0.	0.	0.	0.	0.	0.	9956.	0.0	2021
RAIL SHIPMENTS ■			0.0	2274.2		0.0				
TRUCK SHIPMENTS ■			0.0	2315.3						

TABLE A.1.4. Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 1, MTU
2030 Repository

YEAR	REACTOR DISCHARGE	REACTOR STORAGE	SHIPMENT REACTOR TO AFR	SHIPMENT REACTOR TO REPOSITORY	AFR INVENTORY	SHIPMENT AFR TO REPOSITORY	REPOSITORY			YEAR
							RECEIPTS	INVENTORY	RECEIVING AGE, YEARS	
1980	1163.	7196.	0.	0.	0.	0.	0.	0.	0.0	1980
1981	2760.	9056.	0.	0.	0.	0.	0.	0.	0.0	1981
1982	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	1982
1983	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	1983
1984	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	1984
1985	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	1985
1986	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	1986
1987	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	1987
1988	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	1988
1989	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	1989
1990	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	1990
1991	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	1991
1992	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	1992
1993	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	1993
1994	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	1994
1995	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	1995
1996	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	1996
1997	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	1997
1998	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	1998
1999	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	1999
2000	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2000
2001	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2001
2002	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2002
2003	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2003
2004	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2004
2005	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2005
2006	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2006
2007	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2007
2008	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2008
2009	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2009
2010	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2010
2011	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2011
2012	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2012
2013	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2013
2014	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2014
2015	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2015
2016	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2016
2017	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2017
2018	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2018
2019	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2019
2020	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2020
2021	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2021
2022	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2022
2023	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2023
2024	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2024
2025	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2025
2026	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2026
2027	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2027
2028	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2028
2029	0.	9056.	0.	0.	0.	0.	0.	0.	0.0	2029
2030	0.	9056.	0.	700.	0.	0.	700.	700.	58.0	2030
2031	0.	7056.	0.	1300.	0.	0.	1300.	2000.	58.4	2031
2032	0.	5056.	0.	2000.	0.	0.	2000.	4000.	58.8	2032
2033	0.	3056.	0.	2000.	0.	0.	2000.	6000.	59.2	2033
2034	0.	2760.	0.	1196.	0.	0.	1196.	7196.	59.6	2034
2035	0.	0.	0.	2760.	0.	0.	2760.	9056.	59.9	2035
2036	0.	0.	0.	0.	0.	0.	0.	9056.	60.0	2036
2037	0.	0.	0.	0.	0.	0.	0.	9056.	60.0	2037
2038	0.	0.	0.	0.	0.	0.	0.	9056.	60.0	2038
2039	0.	0.	0.	0.	0.	0.	0.	9056.	60.0	2039
2040	0.	0.	0.	0.	0.	0.	0.	9056.	60.0	2040
2041	0.	0.	0.	0.	0.	0.	0.	9056.	60.0	2041
RAIL SHIPMENTS =			0.0	2274.8		0.0				
TRUCK SHIPMENTS =			0.0	2315.3						

TABLE A.1.5. Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 2, MTU

No Repository

YEAR	REACTOR DISCHARGE	REACTOR STORAGE	SHIPMENT REACTOR TO APP	SHIPMENT REACTOR TO REPOSITORY	APP INVENTORY	SHIPMENT APP TO REPOSITORY	REPOSITORY			YEAR
							RECEIPTS	INVENTORY	RECEIVING AGE, YEARS	
1980	1160.	7196.	0.	0.	0.	0.	0.	0.	0.0	1980
1981	1152.	8149.	0.	0.	0.	0.	0.	0.	0.0	1981
1982	1132.	9481.	0.	0.	0.	0.	0.	0.	0.0	1982
1983	1172.	9704.	949.	0.	949.	0.	0.	0.	0.0	1983
1984	1243.	999A.	949.	0.	1897.	0.	0.	0.	0.0	1984
1985	1157.	10099.	1056.	0.	2953.	0.	0.	0.	0.0	1985
1986	1116.	10099.	1116.	0.	4069.	0.	0.	0.	0.0	1986
1987	1194.	10161.	1132.	0.	5201.	0.	0.	0.	0.0	1987
1988	1224.	10214.	1172.	0.	6373.	0.	0.	0.	0.0	1988
1989	1153.	10226.	1101.	0.	7514.	0.	0.	0.	0.0	1989
1990	1114.	10226.	1114.	0.	862A.	0.	0.	0.	0.0	1990
1991	1170.	10280.	1116.	0.	9704.	0.	0.	0.	0.0	1991
1992	1241.	10327.	1194.	0.	1091A.	0.	0.	0.	0.0	1992
1993	1124.	1031A.	1132.	0.	12071.	0.	0.	0.	0.0	1993
1994	1168.	10333.	1153.	0.	13223.	0.	0.	0.	0.0	1994
1995	1130.	10349.	1114.	0.	14317.	0.	0.	0.	0.0	1995
1996	1224.	10403.	1170.	0.	1550A.	0.	0.	0.	0.0	1996
1997	1196.	10477.	1172.	0.	16644.	0.	0.	0.	0.0	1997
1998	1143.	10491.	1194.	0.	17740.	0.	0.	0.	0.0	1998
1999	1119.	10472.	1138.	0.	18807.	0.	0.	0.	0.0	1999
2000	1193.	10540.	1174.	0.	20021.	0.	0.	0.	0.0	2000
2001	1202.	10574.	1168.	0.	21129.	0.	0.	0.	0.0	2001
2002	1124.	10574.	1174.	0.	22313.	0.	0.	0.	0.0	2002
2003	1040.	10474.	1040.	0.	23342.	0.	0.	0.	0.0	2003
2004	1033.	10574.	1033.	0.	24327.	0.	0.	0.	0.0	2004
2005	1118.	10574.	1118.	0.	25305.	0.	0.	0.	0.0	2005
2006	966.	10474.	966.	0.	26471.	0.	0.	0.	0.0	2006
2007	1010.	10474.	1010.	0.	27441.	0.	0.	0.	0.0	2007
2008	986.	10474.	986.	0.	28447.	0.	0.	0.	0.0	2008
2009	1032.	10474.	1032.	0.	29469.	0.	0.	0.	0.0	2009
2010	1125.	10659.	1040.	0.	30544.	0.	0.	0.	0.0	2010
2011	1066.	10750.	966.	0.	31564.	0.	0.	0.	0.0	2011
2012	1102.	10851.	1010.	0.	32514.	0.	0.	0.	0.0	2012
2013	1086.	10952.	966.	0.	33500.	0.	0.	0.	0.0	2013
2014	1262.	11187.	1032.	0.	34542.	0.	0.	0.	0.0	2014
2015	834.	11263.	963.	0.	35445.	0.	0.	0.	0.0	2015
2016	544.	11263.	544.	0.	36029.	0.	0.	0.	0.0	2016
2017	458.	11263.	458.	0.	36487.	0.	0.	0.	0.0	2017
2018	310.	11263.	310.	0.	36707.	0.	0.	0.	0.0	2018
2019	0.	11263.	0.	0.	36707.	0.	0.	0.	0.0	2019
2020	0.	11263.	0.	0.	36707.	0.	0.	0.	0.0	2020
2021	0.	11263.	0.	0.	36707.	0.	0.	0.	0.0	2021
2022	0.	11263.	0.	0.	36707.	0.	0.	0.	0.0	2022
2023	0.	11263.	0.	0.	36707.	0.	0.	0.	0.0	2023
2024	0.	11263.	0.	0.	36707.	0.	0.	0.	0.0	2024
2025	0.	11263.	0.	0.	36707.	0.	0.	0.	0.0	2025
2026	0.	11263.	0.	0.	36707.	0.	0.	0.	0.0	2026
2027	0.	11263.	0.	0.	36707.	0.	0.	0.	0.0	2027
2028	0.	11263.	0.	0.	36707.	0.	0.	0.	0.0	2028
2029	0.	11263.	0.	0.	36707.	0.	0.	0.	0.0	2029
2030	0.	11263.	0.	0.	36707.	0.	0.	0.	0.0	2030
2031	0.	11263.	0.	0.	36707.	0.	0.	0.	0.0	2031
2032	0.	11263.	0.	0.	36707.	0.	0.	0.	0.0	2032
2033	0.	11263.	0.	0.	36707.	0.	0.	0.	0.0	2033
2034	0.	11263.	0.	0.	36707.	0.	0.	0.	0.0	2034
2035	0.	11263.	0.	0.	36707.	0.	0.	0.	0.0	2035
2036	0.	11263.	0.	0.	36707.	0.	0.	0.	0.0	2036
2037	0.	11263.	0.	0.	36707.	0.	0.	0.	0.0	2037
2038	0.	11263.	0.	0.	36707.	0.	0.	0.	0.0	2038
2039	0.	11263.	0.	0.	36707.	0.	0.	0.	0.0	2039
2040	0.	11263.	0.	0.	36707.	0.	0.	0.	0.0	2040
RAIL SHIPMENTS #			6405.5	0.0		0.0				
TRUCK SHIPMENTS #			8537.6	0.0						

TABLE A.1.6. Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 2, MTU

1990 Repository

YEAR	REACTOR DISCHARGE	REACTOR STORAGE	SHIPMENT REACTOR TO APR	SHIPMENT REACTOR TO REPOSITORY	APR INVENTORY	SHIPMENT APR TO REPOSITORY	REPOSITORY			YEAR
							RECEIPTS	INVENTORY	RECEIVING AGE, YEARS	
1980	1160.	7196.	0.	0.	0.	0.	0.	0.	0.0	1980
1981	1152.	8349.	0.	0.	0.	0.	0.	0.	0.0	1981
1982	1132.	9481.	0.	0.	0.	0.	0.	0.	0.0	1982
1983	1172.	9704.	949.	0.	949.	0.	0.	0.	0.0	1983
1984	1243.	9998.	949.	0.	1897.	0.	0.	0.	0.0	1984
1985	1157.	10099.	1056.	0.	2953.	0.	0.	0.	0.0	1985
1986	1116.	10099.	1116.	0.	4069.	0.	0.	0.	0.0	1986
1987	1194.	10161.	1132.	0.	5261.	0.	0.	0.	0.0	1987
1988	1224.	10214.	1172.	0.	6373.	0.	0.	0.	0.0	1988
1989	1153.	10226.	1141.	0.	7514.	0.	0.	0.	0.0	1989
1990	1114.	10226.	414.	700.	7928.	0.	700.	700.	18.0	1990
1991	1170.	10096.	0.	1300.	7928.	0.	1300.	2000.	16.4	1991
1992	1241.	9543.	0.	1794.	7722.	208.	2000.	4000.	15.3	1992
1993	1124.	10318.	12.	337.	6071.	1663.	2000.	6000.	14.4	1993
1994	1168.	10318.	1030.	137.	5238.	1863.	2000.	8000.	13.7	1994
1995	1130.	10323.	1114.	11.	4363.	1989.	2000.	10000.	12.9	1995
1996	1224.	10390.	988.	170.	3521.	1830.	2000.	12000.	12.2	1996
1997	1196.	9962.	0.	1623.	3164.	377.	2000.	14000.	11.5	1997
1998	1143.	9112.	0.	1994.	3198.	6.	2000.	16000.	10.8	1998
1999	1119.	9527.	0.	704.	1841.	1296.	2000.	18000.	10.1	1999
2000	1193.	10540.	1.	179.	21.	1821.	2000.	20000.	9.4	2000
2001	1202.	9762.	0.	1980.	1.	20.	2000.	22000.	8.7	2001
2002	1124.	8886.	0.	2000.	1.	0.	2000.	24000.	8.0	2002
2003	1040.	7926.	0.	1999.	0.	1.	2000.	26000.	7.2	2003
2004	1035.	6961.	0.	2000.	0.	0.	2000.	28000.	6.6	2004
2005	1118.	6079.	0.	2000.	0.	0.	2000.	30000.	5.8	2005
2006	966.	5982.	0.	1763.	0.	0.	2000.	31763.	5.2	2006
2007	1010.	5168.	0.	1124.	0.	0.	1763.	32887.	5.0	2007
2008	986.	5114.	0.	1040.	0.	0.	1124.	33926.	5.0	2008
2009	1032.	5112.	0.	1035.	0.	0.	1040.	34961.	5.0	2009
2010	1123.	5118.	0.	1118.	0.	0.	1035.	36079.	5.0	2010
2011	1066.	5218.	0.	966.	0.	0.	1118.	37045.	5.0	2011
2012	1102.	5310.	0.	1010.	0.	0.	966.	38054.	5.0	2012
2013	1086.	5410.	0.	986.	0.	0.	1010.	39041.	5.0	2013
2014	1262.	5640.	0.	1032.	0.	0.	986.	40074.	5.0	2014
2015	834.	5350.	0.	1123.	0.	0.	1032.	41198.	5.0	2015
2016	544.	4829.	0.	1066.	0.	0.	1123.	42263.	5.0	2016
2017	458.	4185.	0.	1102.	0.	0.	1066.	43365.	5.0	2017
2018	310.	3409.	0.	1086.	0.	0.	1102.	44431.	5.0	2018
2019	0.	2147.	0.	1262.	0.	0.	1086.	45713.	5.0	2019
2020	0.	1313.	0.	834.	0.	0.	1262.	46548.	5.0	2020
2021	0.	768.	0.	544.	0.	0.	834.	47092.	5.0	2021
2022	0.	310.	0.	458.	0.	0.	544.	47550.	5.0	2022
2023	0.	0.	0.	310.	0.	0.	458.	47860.	5.0	2023
2024	0.	0.	0.	0.	0.	0.	310.	47860.	0.0	2024
2025	0.	0.	0.	0.	0.	0.	0.	47860.	0.0	2025
2026	0.	0.	0.	0.	0.	0.	0.	47860.	0.0	2026
RAIL SHIPMENTS •			2529.3	8403.2		2417.6				
TRUCK SHIPMENTS •			2575.1	8553.2						

TABLE A.1.7. Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 2, MTU.

2010 Repository

YEAR	REACTOR DISCHARGE	REACTOR STORAGE	SHIPMENT REACTOR TO AFR	SHIPMENT REACTOR TO REPOSITORY	AFR INVENTORY	SHIPMENT AFR TO REPOSITORY	REPOSITORY			YEAR
							RECEIPTS	INVENTORY	RECEIVING AGE, YEARS	
1980	1160.	7196.	0.	0.	0.	0.	0.	0.	0.0	1980
1981	1152.	8349.	0.	0.	0.	0.	0.	0.	0.0	1981
1982	1132.	9481.	0.	0.	0.	0.	0.	0.	0.0	1982
1983	1172.	9704.	949.	0.	949.	0.	0.	0.	0.0	1983
1984	1243.	9998.	949.	0.	1897.	0.	0.	0.	0.0	1984
1985	1157.	10099.	1056.	0.	2943.	0.	0.	0.	0.0	1985
1986	1116.	10099.	1116.	0.	4049.	0.	0.	0.	0.0	1986
1987	1194.	10161.	1132.	0.	5241.	0.	0.	0.	0.0	1987
1988	1224.	10214.	1172.	0.	6373.	0.	0.	0.	0.0	1988
1989	1153.	10226.	1141.	0.	7514.	0.	0.	0.	0.0	1989
1990	1114.	10226.	1114.	0.	8628.	0.	0.	0.	0.0	1990
1991	1170.	10280.	1116.	0.	9744.	0.	0.	0.	0.0	1991
1992	1241.	10327.	1194.	0.	10938.	0.	0.	0.	0.0	1992
1993	1124.	10318.	1132.	0.	12071.	0.	0.	0.	0.0	1993
1994	1168.	10333.	1153.	0.	13223.	0.	0.	0.	0.0	1994
1995	1130.	10349.	1114.	0.	14317.	0.	0.	0.	0.0	1995
1996	1224.	10403.	1170.	0.	15508.	0.	0.	0.	0.0	1996
1997	1196.	10472.	1127.	0.	16674.	0.	0.	0.	0.0	1997
1998	1143.	10491.	1124.	0.	17749.	0.	0.	0.	0.0	1998
1999	1119.	10472.	1138.	0.	18897.	0.	0.	0.	0.0	1999
2000	1193.	10540.	1124.	0.	20021.	0.	0.	0.	0.0	2000
2001	1202.	10574.	1168.	0.	21149.	0.	0.	0.	0.0	2001
2002	1124.	10574.	1124.	0.	22313.	0.	0.	0.	0.0	2002
2003	1040.	10574.	1040.	0.	23342.	0.	0.	0.	0.0	2003
2004	1035.	10574.	1035.	0.	24347.	0.	0.	0.	0.0	2004
2005	1118.	10574.	1118.	0.	25345.	0.	0.	0.	0.0	2005
2006	966.	10574.	966.	0.	26471.	0.	0.	0.	0.0	2006
2007	1010.	10574.	1010.	0.	27441.	0.	0.	0.	0.0	2007
2008	966.	10574.	966.	0.	28447.	0.	0.	0.	0.0	2008
2009	1032.	10574.	1032.	0.	29449.	0.	0.	0.	0.0	2009
2010	1125.	10654.	340.	700.	29648.	0.	700.	700.	38.0	2010
2011	1066.	10425.	0.	1300.	29848.	0.	1300.	2000.	36.4	2011
2012	1102.	9733.	0.	1794.	29643.	206.	2000.	4000.	35.3	2012
2013	1086.	10482.	0.	337.	27949.	1663.	2000.	6000.	34.4	2013
2014	1262.	11063.	544.	137.	26641.	1863.	2000.	8000.	33.7	2014
2015	834.	11063.	823.	11.	25445.	1989.	2000.	10000.	32.9	2015
2016	544.	11063.	438.	106.	24049.	1894.	2000.	12000.	32.2	2016
2017	458.	11063.	419.	39.	22447.	1961.	2000.	14000.	31.5	2017
2018	310.	11063.	263.	48.	20747.	1932.	2000.	16000.	30.8	2018
2019	0.	11019.	0.	44.	18841.	1956.	2000.	18000.	30.1	2019
2020	0.	11019.	0.	0.	16841.	2000.	2000.	20000.	29.4	2020
2021	0.	10905.	0.	114.	14945.	1986.	2000.	22000.	28.7	2021
2022	0.	10873.	0.	31.	12947.	1969.	2000.	24000.	28.0	2022
2023	0.	10408.	0.	65.	11042.	1935.	2000.	26000.	27.8	2023
2024	0.	10660.	0.	148.	9240.	1852.	2000.	28000.	26.6	2024
2025	0.	10514.	0.	147.	7346.	1853.	2000.	30000.	25.8	2025
2026	0.	10219.	0.	294.	5641.	1706.	2000.	32000.	25.1	2026
2027	0.	10075.	0.	144.	3745.	1456.	2000.	34000.	24.4	2027
2028	0.	9349.	0.	726.	2511.	1274.	2000.	36000.	23.5	2028
2029	0.	7373.	0.	1976.	2447.	24.	2000.	38000.	22.5	2029
2030	0.	5879.	0.	1494.	1941.	306.	2000.	40000.	21.5	2030
2031	0.	5070.	0.	809.	740.	1191.	2000.	42000.	20.7	2031
2032	0.	3751.	0.	1319.	149.	681.	2000.	44000.	19.8	2032
2033	0.	1860.	0.	1891.	-0.	109.	2000.	46000.	19.1	2033
2034	0.	0.	0.	1860.	-0.	0.	1860.	47860.	17.8	2034
2035	0.	0.	0.	0.	-0.	0.	0.	47860.	0.0	2035
2036	0.	0.	0.	0.	-0.	0.	0.	47860.	0.0	2036
RAIL SHIPMENTS =			7383.9	3548.6		7057.9				
TRUCK SHIPMENTS =			7517.5	3612.6						

TABLE A.1.8. Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 2, MTU
2030 Repository

EAP	REACTOR DISCHARGE	REACTOR STORAGE	SHIPMENT REACTOR TO AFR	SHIPMENT REACTOR TO REPOSITORY	AFR INVENTORY	SHIPMENT AFR TO REPOSITORY	REPOSITORY			YEAR
							RECEIPTS	INVENTORY	RECEIVING AGE, YEARS	
980	1160.	7194.	0.	0.	0.	0.	0.	0.	0.0	1980
981	1152.	8449.	0.	0.	0.	0.	0.	0.	0.0	1981
982	1132.	9481.	0.	0.	0.	0.	0.	0.	0.0	1982
983	1172.	9704.	949.	0.	949.	0.	0.	0.	0.0	1983
984	1243.	9994.	949.	0.	1907.	0.	0.	0.	0.0	1984
985	1157.	10099.	1056.	0.	2963.	0.	0.	0.	0.0	1985
986	1116.	10099.	1116.	0.	4049.	0.	0.	0.	0.0	1986
987	1194.	10161.	1132.	0.	5201.	0.	0.	0.	0.0	1987
988	1224.	10214.	1172.	0.	6373.	0.	0.	0.	0.0	1988
989	1153.	10226.	1141.	0.	7514.	0.	0.	0.	0.0	1989
990	1114.	10226.	1114.	0.	8624.	0.	0.	0.	0.0	1990
991	1170.	10280.	1116.	0.	9744.	0.	0.	0.	0.0	1991
992	1241.	10327.	1194.	0.	10948.	0.	0.	0.	0.0	1992
993	1124.	10318.	1132.	0.	12071.	0.	0.	0.	0.0	1993
994	1168.	10333.	1153.	0.	13223.	0.	0.	0.	0.0	1994
995	1130.	10449.	1114.	0.	14347.	0.	0.	0.	0.0	1995
996	1224.	10403.	1170.	0.	15508.	0.	0.	0.	0.0	1996
997	1196.	10472.	1172.	0.	16674.	0.	0.	0.	0.0	1997
998	1143.	10491.	1124.	0.	17749.	0.	0.	0.	0.0	1998
999	1119.	10472.	1138.	0.	18807.	0.	0.	0.	0.0	1999
000	1193.	10540.	1174.	0.	20021.	0.	0.	0.	0.0	2000
001	1202.	10474.	1168.	0.	21149.	0.	0.	0.	0.0	2001
002	1124.	10574.	1174.	0.	22313.	0.	0.	0.	0.0	2002
003	1040.	10574.	1040.	0.	23342.	0.	0.	0.	0.0	2003
004	1035.	10474.	1035.	0.	24347.	0.	0.	0.	0.0	2004
005	1114.	10574.	1114.	0.	25505.	0.	0.	0.	0.0	2005
006	966.	10474.	966.	0.	26471.	0.	0.	0.	0.0	2006
007	1010.	10474.	1010.	0.	27441.	0.	0.	0.	0.0	2007
008	986.	10474.	986.	0.	28447.	0.	0.	0.	0.0	2008
009	1032.	10474.	1032.	0.	29449.	0.	0.	0.	0.0	2009
010	1125.	10650.	1040.	0.	30544.	0.	0.	0.	0.0	2010
011	1066.	10759.	966.	0.	31504.	0.	0.	0.	0.0	2011
012	1102.	10851.	1010.	0.	32510.	0.	0.	0.	0.0	2012
013	1086.	10952.	986.	0.	33500.	0.	0.	0.	0.0	2013
014	1262.	11182.	1032.	0.	34542.	0.	0.	0.	0.0	2014
015	834.	11063.	966.	0.	35445.	0.	0.	0.	0.0	2015
016	544.	11063.	544.	0.	36029.	0.	0.	0.	0.0	2016
017	458.	11063.	458.	0.	36447.	0.	0.	0.	0.0	2017
018	310.	11063.	310.	0.	36707.	0.	0.	0.	0.0	2018
019	0.	11063.	0.	0.	36707.	0.	0.	0.	0.0	2019
020	0.	11063.	0.	0.	36707.	0.	0.	0.	0.0	2020
021	0.	11063.	0.	0.	36707.	0.	0.	0.	0.0	2021
022	0.	11063.	0.	0.	36707.	0.	0.	0.	0.0	2022
023	0.	11063.	0.	0.	36707.	0.	0.	0.	0.0	2023
024	0.	11063.	0.	0.	36707.	0.	0.	0.	0.0	2024
025	0.	11063.	0.	0.	36707.	0.	0.	0.	0.0	2025
026	0.	11063.	0.	0.	36707.	0.	0.	0.	0.0	2026
027	0.	11063.	0.	0.	36707.	0.	0.	0.	0.0	2027
028	0.	11063.	0.	0.	36707.	0.	0.	0.	0.0	2028
029	0.	11063.	0.	0.	36707.	0.	0.	0.	0.0	2029
030	0.	10363.	0.	700.	36707.	0.	700.	700.	56.0	2030
031	0.	9063.	0.	1300.	36707.	0.	1300.	2000.	56.4	2031
032	0.	7264.	0.	1794.	34502.	204.	2000.	4000.	55.3	2032
033	0.	6032.	0.	337.	34929.	1663.	2000.	6000.	54.4	2033
034	0.	6794.	0.	137.	33046.	1863.	2000.	8000.	53.7	2034
035	0.	6783.	0.	11.	31077.	1949.	2000.	10000.	52.9	2035
036	0.	6677.	0.	106.	29143.	1894.	2000.	12000.	52.2	2036
037	0.	6634.	0.	39.	27222.	1961.	2000.	14000.	51.5	2037
038	0.	6490.	0.	44.	25270.	1952.	2000.	16000.	50.8	2038
039	0.	6546.	0.	44.	23314.	1956.	2000.	18000.	50.1	2039
040	0.	6546.	0.	0.	21314.	2000.	2000.	20000.	49.4	2040
041	0.	6432.	0.	114.	19424.	1886.	2000.	22000.	48.7	2041
042	0.	6401.	0.	31.	17449.	1969.	2000.	24000.	48.0	2042
043	0.	6134.	0.	65.	15524.	1935.	2000.	24000.	47.2	2043
044	0.	6184.	0.	148.	13672.	1952.	2000.	24000.	46.6	2044
045	0.	6441.	0.	147.	11819.	1853.	2000.	24000.	45.8	2045
046	0.	5747.	0.	294.	10113.	1706.	2000.	24000.	45.1	2046
047	0.	5603.	0.	144.	8247.	1856.	2000.	24000.	44.4	2047
048	0.	5427.	0.	76.	6373.	1924.	2000.	24000.	43.5	2048
049	0.	5522.	0.	6.	4349.	1994.	2000.	24000.	42.5	2049
050	0.	5622.	0.	0.	2349.	2000.	2000.	24000.	41.5	2050
051	0.	4057.	0.	564.	923.	1436.	2000.	22000.	40.7	2051
052	0.	3731.	0.	1226.	129.	774.	2000.	20000.	39.8	2052
053	0.	1460.	0.	1871.	0.	129.	2000.	16000.	39.1	2053
054	0.	0.	0.	1860.	0.	0.	1960.	47860.	37.8	2054
055	0.	0.	0.	0.	0.	0.	0.	47860.	0.0	2055
056	0.	0.	0.	0.	0.	0.	0.	47860.	0.0	2056
ALL SHIPMENTS =			8405.5	2527.0		8014.4				
RUFK SHIPMENTS =			5557.6	2527.7						

TABLE A.1.9. Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 3, MTU

No Repository

YEAR	REACTOR DISCHARGE	REACTOR RYORAGE	SHIPMENT REACTOR TO AFR	SHIPMENT REACTOR TO REPOSITORY	AFR INVENTORY	SHIPMENT AFR TO REPOSITORY	REPOSITORY		RECEIVING AGE, YEARS	YEAR
							RECEIPTS	INVENTORY		
1980	1160.	7196.	0.	0.	0.	0.	0.	0.	0.0	1980
1981	1282.	8078.	0.	0.	0.	0.	0.	0.	0.0	1981
1982	1486.	9060.	0.	0.	0.	0.	0.	0.	0.0	1982
1983	1770.	10788.	0.	0.	0.	0.	0.	0.	0.0	1983
1984	2150.	11991.	0.	0.	1807.	0.	0.	0.	0.0	1984
1985	2333.	13271.	0.	0.	2953.	0.	0.	0.	0.0	1985
1986	2605.	14666.	0.	0.	4143.	0.	0.	0.	0.0	1986
1987	2830.	16336.	0.	0.	5323.	0.	0.	0.	0.0	1987
1988	3045.	18099.	0.	0.	6604.	0.	0.	0.	0.0	1988
1989	3159.	19772.	0.	0.	8000.	0.	0.	0.	0.0	1989
1990	3368.	21370.	0.	0.	9841.	0.	0.	0.	0.0	1990
1991	3646.	22867.	0.	0.	12015.	0.	0.	0.	0.0	1991
1992	3858.	24384.	0.	0.	14360.	0.	0.	0.	0.0	1992
1993	3938.	25717.	0.	0.	16905.	0.	0.	0.	0.0	1993
1994	4235.	27121.	0.	0.	19705.	0.	0.	0.	0.0	1994
1995	4380.	28457.	0.	0.	22870.	0.	0.	0.	0.0	1995
1996	4588.	29844.	0.	0.	25909.	0.	0.	0.	0.0	1996
1997	4854.	31371.	0.	0.	29327.	0.	0.	0.	0.0	1997
1998	5083.	32808.	0.	0.	33003.	0.	0.	0.	0.0	1998
1999	5303.	34250.	0.	0.	36858.	0.	0.	0.	0.0	1999
2000	5595.	35915.	0.	0.	40708.	0.	0.	0.	0.0	2000
2001	5894.	37530.	0.	0.	45073.	0.	0.	0.	0.0	2001
2002	5752.	38904.	0.	0.	49413.	0.	0.	0.	0.0	2002
2003	5749.	40067.	0.	0.	54001.	0.	0.	0.	0.0	2003
2004	5692.	40003.	0.	0.	58824.	0.	0.	0.	0.0	2004
2005	5766.	41509.	0.	0.	63977.	0.	0.	0.	0.0	2005
2006	5627.	41012.	0.	0.	69200.	0.	0.	0.	0.0	2006
2007	5727.	42040.	0.	0.	74809.	0.	0.	0.	0.0	2007
2008	5647.	42040.	0.	0.	80484.	0.	0.	0.	0.0	2008
2009	5725.	42073.	0.	0.	86178.	0.	0.	0.	0.0	2009
2010	5788.	42095.	0.	0.	91904.	0.	0.	0.	0.0	2010
2011	5631.	42100.	0.	0.	97570.	0.	0.	0.	0.0	2011
2012	5700.	42170.	0.	0.	103201.	0.	0.	0.	0.0	2012
2013	5669.	42172.	0.	0.	108914.	0.	0.	0.	0.0	2013
2014	5796.	42193.	0.	0.	114803.	0.	0.	0.	0.0	2014
2015	5182.	42191.	0.	0.	119827.	0.	0.	0.	0.0	2015
2016	4993.	42191.	0.	0.	124920.	0.	0.	0.	0.0	2016
2017	4858.	42191.	0.	0.	129694.	0.	0.	0.	0.0	2017
2018	4570.	42191.	0.	0.	134208.	0.	0.	0.	0.0	2018
2019	4313.	42191.	0.	0.	138369.	0.	0.	0.	0.0	2019
2020	4655.	42191.	0.	0.	143214.	0.	0.	0.	0.0	2020
2021	4786.	42191.	0.	0.	148000.	0.	0.	0.	0.0	2021
2022	4617.	42191.	0.	0.	152617.	0.	0.	0.	0.0	2022
2023	4543.	42191.	0.	0.	157140.	0.	0.	0.	0.0	2023
2024	4150.	42191.	0.	0.	161310.	0.	0.	0.	0.0	2024
2025	3789.	42191.	0.	0.	165009.	0.	0.	0.	0.0	2025
2026	3374.	42191.	0.	0.	168474.	0.	0.	0.	0.0	2026
2027	3224.	42191.	0.	0.	171608.	0.	0.	0.	0.0	2027
2028	2948.	42191.	0.	0.	174607.	0.	0.	0.	0.0	2028
2029	2813.	42191.	0.	0.	177460.	0.	0.	0.	0.0	2029
2030	2598.	42191.	0.	0.	180008.	0.	0.	0.	0.0	2030
2031	2353.	42191.	0.	0.	182470.	0.	0.	0.	0.0	2031
2032	2268.	42191.	0.	0.	184678.	0.	0.	0.	0.0	2032
2033	2033.	42191.	0.	0.	186711.	0.	0.	0.	0.0	2033
2034	1818.	42191.	0.	0.	188529.	0.	0.	0.	0.0	2034
2035	1732.	42191.	0.	0.	190241.	0.	0.	0.	0.0	2035
2036	1810.	42191.	0.	0.	192071.	0.	0.	0.	0.0	2036
2037	1519.	42191.	0.	0.	193500.	0.	0.	0.	0.0	2037
2038	1263.	42191.	0.	0.	194845.	0.	0.	0.	0.0	2038
2039	1194.	42191.	0.	0.	196009.	0.	0.	0.	0.0	2039
2040	752.	42191.	0.	0.	196801.	0.	0.	0.	0.0	2040
2041	0.	42191.	0.	0.	196801.	0.	0.	0.	0.0	2041
RAIL SHIPMENTS =			44950.5	0.0						
TRUCK SHIPMENTS =			43767.6	0.0						

TABLE A.1.10. Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 3, MTU

1990 Repository

YEAR	REACTOR DISCHARGE	REACTOR STORAGE	SHIPMENT REACTOR TO AFR	SHIPMENT REACTOR TO REPOSITORY	AFR INVENTORY	SHIPMENT AFR TO REPOSITORY	REPOSITORY			YEAR
							RECEIPTS	INVENTORY	RECEIVING AGE, YEARS	
1980	1160.	7196.	0.	0.	0.	0.	0.	0.	0.0	1980
1981	1242.	8474.	0.	0.	0.	0.	0.	0.	0.0	1981
1982	1486.	9664.	0.	0.	0.	0.	0.	0.	0.0	1982
1983	1770.	10786.	949.	0.	949.	0.	0.	0.	0.0	1983
1984	2154.	11991.	949.	0.	1897.	0.	0.	0.	0.0	1984
1985	2335.	13271.	1056.	0.	2953.	0.	0.	0.	0.0	1985
1986	2605.	14666.	1209.	0.	4163.	0.	0.	0.	0.0	1986
1987	2830.	16336.	1160.	0.	5323.	0.	0.	0.	0.0	1987
1988	3045.	18099.	1242.	0.	6604.	0.	0.	0.	0.0	1988
1989	3159.	19772.	1446.	0.	8000.	0.	0.	0.	0.0	1989
1990	3368.	21370.	1070.	700.	9161.	0.	700.	700.	18.0	1990
1991	3646.	22862.	854.	1300.	10015.	0.	1300.	2000.	16.4	1991
1992	3858.	24384.	540.	1794.	10340.	206.	2000.	4000.	15.3	1992
1993	3938.	25717.	2268.	337.	10945.	1663.	2000.	6000.	14.4	1993
1994	4235.	27121.	2673.	157.	11745.	1843.	2000.	8000.	13.7	1994
1995	4380.	28457.	2489.	556.	12170.	2144.	2700.	10700.	12.9	1995
1996	4584.	29886.	2145.	1014.	11949.	2286.	4300.	14000.	12.2	1996
1997	4854.	31371.	970.	2398.	11347.	1602.	4000.	18000.	11.6	1997
1998	5083.	32808.	712.	2934.	11043.	1066.	4000.	22000.	11.1	1998
1999	5303.	34254.	2968.	889.	10840.	3111.	4000.	25000.	10.7	1999
2000	5599.	35915.	2455.	1443.	8768.	4000.	4000.	32000.	10.2	2000
2001	5854.	36660.	0.	5108.	5946.	2892.	4000.	40000.	9.2	2001
2002	5752.	38477.	0.	3936.	1842.	4064.	4000.	48000.	8.2	2002
2003	5749.	38068.	0.	6158.	-0.	1542.	4000.	56000.	7.3	2003
2004	5692.	35760.	0.	4000.	-0.	0.	4000.	64000.	6.7	2004
2005	5766.	33426.	0.	8000.	-0.	0.	4000.	72000.	6.1	2005
2006	5627.	31152.	0.	4000.	-0.	0.	4000.	80000.	5.6	2006
2007	5727.	28879.	0.	4000.	-0.	0.	4000.	88000.	5.3	2007
2008	5647.	28458.	0.	6068.	-0.	0.	4068.	94068.	5.0	2008
2009	5725.	28491.	0.	5692.	-0.	0.	4692.	99760.	5.0	2009
2010	5788.	28514.	0.	5766.	-0.	0.	4766.	105526.	5.0	2010
2011	5631.	28518.	0.	5627.	-0.	0.	4627.	111152.	5.0	2011
2012	5700.	28491.	0.	5727.	-0.	0.	4727.	116879.	5.0	2012
2013	5669.	28514.	0.	5647.	-0.	0.	4647.	122526.	5.0	2013
2014	5796.	28584.	0.	5725.	-0.	0.	4725.	128251.	5.0	2014
2015	5182.	27479.	0.	5788.	-0.	0.	4788.	134039.	5.0	2015
2016	4993.	27341.	0.	5631.	-0.	0.	4631.	139670.	5.0	2016
2017	4856.	26497.	0.	5700.	-0.	0.	4700.	145370.	5.0	2017
2018	4570.	25997.	0.	5669.	-0.	0.	4669.	151039.	5.0	2018
2019	4313.	23914.	0.	5796.	-0.	0.	4796.	156836.	5.0	2019
2020	4655.	23384.	0.	5182.	-0.	0.	4182.	162018.	5.0	2020
2021	4786.	23180.	0.	4993.	-0.	0.	4993.	167011.	5.0	2021
2022	4817.	22941.	0.	4856.	-0.	0.	4856.	171867.	5.0	2022
2023	4543.	22914.	0.	4570.	-0.	0.	4570.	176437.	5.0	2023
2024	4150.	22751.	0.	4313.	-0.	0.	4313.	180759.	5.0	2024
2025	3789.	21889.	0.	4655.	-0.	0.	4655.	185409.	5.0	2025
2026	3375.	20474.	0.	4786.	-0.	0.	4786.	190192.	5.0	2026
2027	3224.	19081.	0.	4617.	-0.	0.	4617.	194608.	5.0	2027
2028	2948.	17487.	0.	4543.	-0.	0.	4543.	199351.	5.0	2028
2029	2813.	16150.	0.	4150.	-0.	0.	4150.	203501.	5.0	2029
2030	2598.	14454.	0.	3789.	-0.	0.	3789.	207290.	5.0	2030
2031	2353.	13036.	0.	3375.	-0.	0.	3375.	210669.	5.0	2031
2032	2268.	12980.	0.	3224.	-0.	0.	3224.	213488.	5.0	2032
2033	2033.	12064.	0.	2948.	-0.	0.	2948.	216434.	5.0	2033
2034	1818.	11070.	0.	2813.	-0.	0.	2813.	219651.	5.0	2034
2035	1732.	10203.	0.	2598.	-0.	0.	2598.	222244.	5.0	2035
2036	1810.	9661.	0.	2353.	-0.	0.	2353.	224601.	5.0	2036
2037	1519.	8912.	0.	2268.	-0.	0.	2268.	226869.	5.0	2037
2038	1269.	8144.	0.	2033.	-0.	0.	2033.	228402.	5.0	2038
2039	1194.	7419.	0.	1818.	-0.	0.	1818.	230721.	5.0	2039
2040	752.	6540.	0.	1732.	-0.	0.	1732.	232452.	5.0	2040
2041	0.	4729.	0.	1810.	-0.	0.	1810.	234263.	5.0	2041
2042	0.	3211.	0.	1519.	-0.	0.	1519.	235781.	5.0	2042
2043	0.	1446.	0.	1269.	-0.	0.	1269.	237046.	5.0	2043
2044	0.	752.	0.	1194.	-0.	0.	1194.	238240.	5.0	2044
2045	0.	0.	0.	752.	-0.	0.	752.	238992.	5.0	2045
2046	0.	0.	0.	0.	-0.	0.	0.	238992.	0.0	2046
2047	0.	0.	0.	0.	-0.	0.	0.	238992.	0.0	2047
2048	0.	0.	0.	0.	-0.	0.	0.	238992.	0.0	2048
2049	0.	0.	0.	0.	-0.	0.	0.	238992.	0.0	2049
2050	0.	0.	0.	0.	-0.	0.	0.	238992.	0.0	2050
2051	0.	0.	0.	0.	-0.	0.	0.	238992.	0.0	2051
RAIL SHIPMENTS =			6221.3	48370.7		5946.6				
TRUCK SHIPMENTS =			6333.9	49245.7						

TABLE A.1.11. Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 3, MTU
2010 Repository

YEAR	REACTOR DISCHARGE	REACTOR STORAGE	SHIPMENT REACTOR TO AFR	SHIPMENT REACTOR TO REPOSITORY	AFR INVENTORY	SHIPMENT AFR TO REPOSITORY	REPOSITORY			RECEIVING AGE, YEARS	YEAR
							RECEIPTS	INVENTORY			
1980	1160.	7196.	0.	0.	0.	0.	0.	0.	0.	0.0	1980
1981	1282.	8478.	0.	0.	0.	0.	0.	0.	0.	0.0	1981
1982	1466.	9964.	0.	0.	0.	0.	0.	0.	0.	0.0	1982
1983	1770.	10786.	949.	0.	949.	0.	0.	0.	0.	0.0	1983
1984	2154.	11991.	949.	0.	1897.	0.	0.	0.	0.	0.0	1984
1985	2335.	13271.	1056.	0.	2953.	0.	0.	0.	0.	0.0	1985
1986	2605.	14666.	1209.	0.	4163.	0.	0.	0.	0.	0.0	1986
1987	2830.	16336.	1160.	0.	5323.	0.	0.	0.	0.	0.0	1987
1988	3045.	18099.	1282.	0.	6604.	0.	0.	0.	0.	0.0	1988
1989	3159.	19772.	1486.	0.	8060.	0.	0.	0.	0.	0.0	1989
1990	3368.	21370.	1770.	0.	9841.	0.	0.	0.	0.	0.0	1990
1991	3646.	22862.	2154.	0.	12015.	0.	0.	0.	0.	0.0	1991
1992	3658.	24384.	2335.	0.	14350.	0.	0.	0.	0.	0.0	1992
1993	3938.	25717.	2605.	0.	16955.	0.	0.	0.	0.	0.0	1993
1994	4233.	27121.	2830.	0.	19745.	0.	0.	0.	0.	0.0	1994
1995	4380.	28457.	3045.	0.	22870.	0.	0.	0.	0.	0.0	1995
1996	4588.	29886.	3159.	0.	25949.	0.	0.	0.	0.	0.0	1996
1997	4854.	31371.	3368.	0.	29347.	0.	0.	0.	0.	0.0	1997
1998	5083.	32808.	3646.	0.	33083.	0.	0.	0.	0.	0.0	1998
1999	5303.	34254.	3456.	0.	36840.	0.	0.	0.	0.	0.0	1999
2000	5399.	35615.	3938.	0.	40768.	0.	0.	0.	0.	0.0	2000
2001	5854.	37534.	4233.	0.	45033.	0.	0.	0.	0.	0.0	2001
2002	5752.	38906.	4380.	0.	49413.	0.	0.	0.	0.	0.0	2002
2003	5749.	40067.	4588.	0.	54001.	0.	0.	0.	0.	0.0	2003
2004	5692.	40405.	4854.	0.	58844.	0.	0.	0.	0.	0.0	2004
2005	5766.	41549.	5083.	0.	63977.	0.	0.	0.	0.	0.0	2005
2006	5627.	41912.	5303.	0.	69240.	0.	0.	0.	0.	0.0	2006
2007	5727.	42040.	5599.	0.	74849.	0.	0.	0.	0.	0.0	2007
2008	5647.	42040.	5647.	0.	80486.	0.	0.	0.	0.	0.0	2008
2009	5725.	42073.	5642.	0.	86178.	0.	0.	0.	0.	0.0	2009
2010	5788.	42079.	5082.	700.	91240.	0.	700.	700.	36.0	2010	
2011	5631.	42079.	4331.	1300.	95501.	0.	1300.	2000.	36.4	2011	
2012	5700.	42079.	3906.	1794.	99241.	206.	2000.	4000.	33.3	2012	
2013	5669.	42122.	5240.	337.	102918.	1663.	2000.	6000.	34.4	2013	
2014	5796.	42191.	5569.	157.	106644.	1843.	2000.	8000.	33.7	2014	
2015	5182.	42191.	4626.	556.	109127.	2144.	2700.	10700.	32.9	2015	
2016	4993.	42191.	4013.	981.	110620.	2319.	3300.	14000.	32.2	2016	
2017	4856.	42191.	4011.	885.	111676.	3155.	4000.	18000.	31.6	2017	
2018	4570.	42191.	3882.	688.	112286.	3312.	4000.	22000.	31.1	2018	
2019	4313.	42191.	3788.	526.	112599.	3474.	4000.	26000.	30.7	2019	
2020	4659.	42191.	4036.	620.	112124.	3380.	6000.	30000.	30.2	2020	
2021	4786.	42191.	3819.	967.	108080.	7933.	4000.	40000.	29.2	2021	
2022	4617.	42191.	3729.	888.	102617.	9112.	10000.	50000.	28.0	2022	
2023	4543.	42191.	3462.	1081.	95160.	10919.	12000.	62000.	26.5	2023	
2024	4150.	42191.	2879.	1271.	85310.	12729.	14000.	76000.	25.9	2024	
2025	3759.	42191.	2990.	799.	75099.	13201.	16000.	90000.	25.4	2025	
2026	3375.	42191.	2800.	575.	64474.	13425.	14000.	104000.	22.0	2026	
2027	3224.	42047.	0.	3366.	53843.	10632.	14000.	118000.	20.5	2027	
2028	2948.	42191.	1609.	1195.	42647.	12405.	14000.	132000.	19.1	2028	
2029	2813.	41089.	0.	3916.	32542.	10684.	14000.	146000.	17.6	2029	
2030	2598.	39484.	0.	4298.	22840.	9702.	14000.	160000.	16.2	2030	
2031	2353.	38477.	0.	3265.	12124.	10735.	14000.	174000.	14.5	2031	
2032	2268.	35978.	0.	4767.	2841.	9233.	14000.	188000.	12.4	2032	
2033	2033.	26902.	0.	11109.	-.	2491.	14000.	202000.	10.5	2033	
2034	1816.	14721.	0.	14000.	-.	0.	14000.	216000.	8.0	2034	
2035	1732.	10203.	0.	6249.	-.	0.	4249.	222240.	5.7	2035	
2036	1810.	9661.	0.	2353.	-.	0.	2353.	224601.	5.0	2036	
2037	1519.	8912.	0.	2268.	-.	0.	2268.	226869.	5.0	2037	
2038	1265.	8144.	0.	2033.	-.	0.	2033.	228402.	5.0	2038	
2039	1194.	7419.	0.	1816.	-.	0.	1816.	230721.	5.0	2039	
2040	752.	6540.	0.	1732.	-.	0.	1732.	232459.	5.0	2040	
2041	0.	4729.	0.	1810.	-.	0.	1810.	234263.	5.0	2041	
2042	0.	3211.	0.	1519.	-.	0.	1519.	235781.	5.0	2042	
2043	0.	1946.	0.	1265.	-.	0.	1265.	237046.	5.0	2043	
2044	0.	752.	0.	1194.	-.	0.	1194.	238240.	5.0	2044	
2045	0.	0.	0.	752.	-.	0.	752.	238992.	5.0	2045	
2046	0.	0.	0.	0.	-.	0.	0.	238992.	0.0	2046	
2047	0.	0.	0.	0.	-.	0.	0.	238992.	0.0	2047	
2048	0.	0.	0.	0.	-.	0.	0.	238992.	0.0	2048	
2049	0.	0.	0.	0.	-.	0.	0.	238992.	0.0	2049	
2050	0.	0.	0.	0.	-.	0.	0.	238992.	0.0	2050	
2051	0.	0.	0.	0.	-.	0.	0.	238992.	0.0	2051	
RAIL SHIPMENTS =			35634.2	18957.8		34060.9					
TRUCK SHIPMENTS =			36274.8	19300.7							

TABLE A.1.12. Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 3, MTU
2030 Repository

YEAR	REACTOR DISCHARGE	REACTOR STORAGE	SHIPMENT REACTOR TO AFR	SHIPMENT REACTOR TO REPOSITORY	AFR INVENTORY	SHIPMENT AFR TO REPOSITORY	REPOSITORY		RECEIVING AGE, YEARS	YEAR
							RECEIPTS	INVENTORY		
1980	1160.	7196.	0.	0.	0.	0.	0.	0.	0.0	1980
1981	1282.	8478.	0.	0.	0.	0.	0.	0.	0.0	1981
1982	1486.	9964.	0.	0.	0.	0.	0.	0.	0.0	1982
1983	1770.	10786.	949.	0.	949.	0.	0.	0.	0.0	1983
1984	2154.	11991.	949.	0.	1897.	0.	0.	0.	0.0	1984
1985	2335.	13271.	1056.	0.	2953.	0.	0.	0.	0.0	1985
1986	2605.	14666.	1209.	0.	4163.	0.	0.	0.	0.0	1986
1987	2830.	16336.	1160.	0.	5323.	0.	0.	0.	0.0	1987
1988	3045.	18099.	1282.	0.	6604.	0.	0.	0.	0.0	1988
1989	3159.	19772.	1486.	0.	8000.	0.	0.	0.	0.0	1989
1990	3368.	21370.	1770.	0.	9861.	0.	0.	0.	0.0	1990
1991	3646.	22867.	2154.	0.	12015.	0.	0.	0.	0.0	1991
1992	3954.	24364.	2335.	0.	14350.	0.	0.	0.	0.0	1992
1993	3938.	25717.	2605.	0.	16955.	0.	0.	0.	0.0	1993
1994	4235.	27121.	2830.	0.	19745.	0.	0.	0.	0.0	1994
1995	4380.	28457.	3045.	0.	22810.	0.	0.	0.	0.0	1995
1996	4588.	29808.	3159.	0.	25949.	0.	0.	0.	0.0	1996
1997	4854.	31371.	3368.	0.	29347.	0.	0.	0.	0.0	1997
1998	5083.	32808.	3646.	0.	33083.	0.	0.	0.	0.0	1998
1999	5303.	34254.	3858.	0.	36840.	0.	0.	0.	0.0	1999
2000	5599.	35915.	3938.	0.	40708.	0.	0.	0.	0.0	2000
2001	5854.	37534.	4235.	0.	45073.	0.	0.	0.	0.0	2001
2002	5752.	38906.	4380.	0.	49413.	0.	0.	0.	0.0	2002
2003	5749.	40067.	4588.	0.	54011.	0.	0.	0.	0.0	2003
2004	5692.	40905.	4854.	0.	59854.	0.	0.	0.	0.0	2004
2005	5766.	41589.	5083.	0.	63917.	0.	0.	0.	0.0	2005
2006	5627.	41917.	5303.	0.	69200.	0.	0.	0.	0.0	2006
2007	5727.	42040.	5599.	0.	74839.	0.	0.	0.	0.0	2007
2008	5647.	42040.	5607.	0.	80486.	0.	0.	0.	0.0	2008
2009	5725.	42073.	5692.	0.	86178.	0.	0.	0.	0.0	2009
2010	5788.	42095.	5766.	0.	91904.	0.	0.	0.	0.0	2010
2011	5631.	42100.	5627.	0.	97570.	0.	0.	0.	0.0	2011
2012	5700.	42079.	5721.	0.	103201.	0.	0.	0.	0.0	2012
2013	5669.	42122.	5627.	0.	108918.	0.	0.	0.	0.0	2013
2014	5796.	42193.	5725.	0.	114643.	0.	0.	0.	0.0	2014
2015	5182.	42191.	5184.	0.	119827.	0.	0.	0.	0.0	2015
2016	4993.	42191.	4993.	0.	124820.	0.	0.	0.	0.0	2016
2017	4856.	42191.	4856.	0.	129676.	0.	0.	0.	0.0	2017
2018	4570.	42191.	4570.	0.	134286.	0.	0.	0.	0.0	2018
2019	4313.	42191.	4313.	0.	138549.	0.	0.	0.	0.0	2019
2020	4655.	42191.	4655.	0.	142214.	0.	0.	0.	0.0	2020
2021	4786.	42191.	4786.	0.	144000.	0.	0.	0.	0.0	2021
2022	4617.	42191.	4617.	0.	152617.	0.	0.	0.	0.0	2022
2023	4503.	42191.	4503.	0.	157140.	0.	0.	0.	0.0	2023
2024	4150.	42191.	4150.	0.	161370.	0.	0.	0.	0.0	2024
2025	3789.	42191.	3789.	0.	165099.	0.	0.	0.	0.0	2025
2026	3375.	42191.	3375.	0.	168474.	0.	0.	0.	0.0	2026
2027	3224.	42191.	3224.	0.	171698.	0.	0.	0.	0.0	2027
2028	2948.	42191.	2948.	0.	174807.	0.	0.	0.	0.0	2028
2029	2813.	42191.	2813.	0.	177440.	0.	0.	0.	0.0	2029
2030	2598.	42191.	1898.	700.	179388.	0.	700.	700.	58.0	2030
2031	2353.	42191.	1053.	1300.	180410.	0.	1300.	2000.	56.4	2031
2032	2268.	42191.	474.	1794.	180678.	206.	2000.	4000.	55.3	2032
2033	2033.	42191.	1696.	337.	180711.	1663.	2000.	6000.	54.4	2033
2034	1818.	42191.	1661.	157.	180929.	1843.	2000.	4000.	53.7	2034
2035	1732.	42191.	1176.	556.	179541.	2144.	2700.	10700.	52.9	2035
2036	1810.	42191.	830.	981.	178071.	2319.	4300.	14000.	52.2	2036
2037	1519.	42191.	674.	845.	175560.	3155.	4000.	18000.	51.6	2037
2038	1265.	42191.	577.	688.	172845.	3312.	4000.	22000.	51.1	2038
2039	1194.	42191.	668.	526.	170029.	3476.	4000.	26000.	50.7	2039
2040	752.	42191.	132.	620.	164801.	5387.	4000.	32000.	50.2	2040
2041	0.	41224.	0.	967.	157748.	7033.	4000.	40000.	49.2	2041
2042	0.	40336.	0.	888.	148646.	9117.	14000.	40000.	48.0	2042
2043	0.	39256.	0.	1081.	137746.	10919.	12000.	42000.	46.5	2043
2044	0.	37985.	0.	1271.	125007.	12729.	14000.	74000.	44.9	2044
2045	0.	37185.	0.	799.	111807.	13201.	14000.	90000.	43.4	2045
2046	0.	37113.	0.	72.	97879.	13924.	14000.	104000.	42.0	2046
2047	0.	37103.	0.	10.	83849.	13990.	14000.	118000.	40.5	2047
2048	0.	36694.	0.	408.	70296.	13592.	14000.	132000.	39.1	2048
2049	0.	34878.	0.	1817.	58114.	12183.	14000.	146000.	37.6	2049
2050	0.	32096.	0.	2782.	46806.	11218.	14000.	160000.	36.2	2050
2051	0.	31033.	0.	664.	33549.	13336.	14000.	174000.	34.5	2051
2052	0.	29624.	0.	1809.	21388.	12191.	14000.	188000.	32.4	2052
2053	0.	25137.	0.	4087.	11845.	9513.	14000.	202000.	30.5	2053
2054	0.	16793.	0.	8344.	6109.	5656.	14000.	216000.	28.0	2054
2055	0.	8498.	0.	8198.	307.	5802.	14000.	230000.	24.2	2055
2056	0.	0.	0.	4595.	-0.	397.	4992.	238992.	19.3	2056
2057	0.	0.	0.	0.	-0.	0.	0.	238992.	0.0	2057
2058	0.	0.	0.	0.	-0.	0.	0.	238992.	0.0	2058
2059	0.	0.	0.	0.	-0.	0.	0.	238992.	0.0	2059
2060	0.	0.	0.	0.	-0.	0.	0.	238992.	0.0	2060
2061	0.	0.	0.	0.	-0.	0.	0.	238992.	0.0	2061
RAIL SHIPMENTS *			43012.0	11580.1		41112.9				
TRUCK SHIPMENTS *			43790.0	11789.5						

TABLE A.1.13. Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 4, MTU
2000 Repository

YEAR	REACTOR DISCHARGE	REACTOR STORAGE	SHIPMENT REACTOR TO AFR	SHIPMENT REACTOR TO REPOSITORY	AFR INVENTORY	SHIPMENT AFR TO REPOSITORY	REPOSITORY			YEAR
							RECEIPTS	INVENTORY	RECEIVING AGE, YEARS	
1980	1160.	7196.	0.	0.	0.	0.	0.	0.	0.0	1980
1981	1282.	8478.	0.	0.	0.	0.	0.	0.	0.0	1981
1982	1486.	9664.	0.	0.	0.	0.	0.	0.	0.0	1982
1983	1770.	10786.	949.	0.	949.	0.	0.	0.	0.0	1983
1984	2154.	11491.	949.	0.	1897.	0.	0.	0.	0.0	1984
1985	2333.	13271.	1056.	0.	2943.	0.	0.	0.	0.0	1985
1986	2605.	14666.	1209.	0.	4143.	0.	0.	0.	0.0	1986
1987	2830.	16336.	1160.	0.	5353.	0.	0.	0.	0.0	1987
1988	3045.	18099.	1282.	0.	6664.	0.	0.	0.	0.0	1988
1989	3159.	19772.	1486.	0.	8000.	0.	0.	0.	0.0	1989
1990	3368.	21370.	1770.	0.	9841.	0.	0.	0.	0.0	1990
1991	3606.	22862.	2154.	0.	12015.	0.	0.	0.	0.0	1991
1992	3858.	24384.	2333.	0.	14340.	0.	0.	0.	0.0	1992
1993	3938.	25717.	2605.	0.	16945.	0.	0.	0.	0.0	1993
1994	4235.	27121.	2830.	0.	19745.	0.	0.	0.	0.0	1994
1995	4380.	28457.	3045.	0.	22840.	0.	0.	0.	0.0	1995
1996	4588.	29880.	3159.	0.	25949.	0.	0.	0.	0.0	1996
1997	4854.	31371.	3368.	0.	29347.	0.	0.	0.	0.0	1997
1998	5083.	32808.	3606.	0.	33043.	0.	0.	0.	0.0	1998
1999	5303.	34254.	3858.	0.	36860.	0.	0.	0.	0.0	1999
2000	5599.	35919.	3238.	700.	40048.	0.	700.	700.	25.0	2000
2001	5854.	37534.	2935.	1300.	43033.	0.	1300.	2000.	25.4	2001
2002	5757.	38911.	2586.	1794.	45413.	206.	2000.	4000.	25.3	2002
2003	5761.	40083.	4251.	337.	48011.	1663.	2000.	6000.	25.4	2003
2004	5705.	40934.	4696.	157.	50844.	1443.	2000.	8000.	23.7	2004
2005	5779.	41631.	4527.	556.	53237.	2144.	2700.	10700.	22.9	2005
2006	5639.	41967.	4323.	981.	55240.	2319.	3300.	14000.	22.2	2006
2007	5740.	42107.	4754.	845.	56849.	3155.	4000.	18000.	21.6	2007
2008	5661.	42107.	4973.	688.	58540.	3312.	4000.	22000.	21.1	2008
2009	5763.	42114.	5231.	528.	60247.	3074.	4000.	26000.	20.7	2009
2010	5850.	42203.	5121.	620.	60017.	5360.	4000.	32000.	20.2	2010
2011	5761.	42260.	4737.	967.	57722.	7033.	4000.	40000.	19.2	2011
2012	5885.	42366.	4397.	1382.	53511.	8618.	10000.	50000.	18.0	2012
2013	5995.	42722.	1174.	4465.	47140.	7535.	12000.	62000.	16.5	2013
2014	6263.	43245.	3956.	1783.	40840.	10217.	12000.	74000.	15.1	2014
2015	5866.	43452.	3012.	2649.	34540.	9351.	12000.	86000.	13.9	2015
2016	5844.	43532.	4245.	1518.	28344.	10482.	12000.	98000.	12.9	2016
2017	5797.	43532.	4302.	1495.	22101.	10503.	12000.	110000.	11.8	2017
2018	5618.	43532.	78.	5541.	15719.	6459.	12000.	122000.	10.7	2018
2019	5520.	43532.	501.	5019.	9249.	6981.	12000.	134000.	9.6	2019
2020	5810.	43532.	2472.	3338.	3049.	8662.	12000.	146000.	8.5	2020
2021	6064.	38163.	0.	11433.	2442.	567.	12000.	158000.	7.5	2021
2022	6094.	34739.	0.	9518.	-0.	2482.	12000.	170000.	6.5	2022
2023	6309.	29796.	0.	11252.	-0.	0.	12000.	181252.	5.5	2023
2024	6300.	30577.	0.	5520.	-0.	0.	5520.	186771.	5.0	2024
2025	6120.	30686.	0.	5610.	-0.	0.	5610.	192581.	5.0	2025
2026	5973.	30795.	0.	6064.	-0.	0.	4064.	198645.	5.0	2026
2027	6047.	30749.	0.	6094.	-0.	0.	4094.	204739.	5.0	2027
2028	5990.	30430.	0.	6309.	-0.	0.	6309.	211048.	5.0	2028
2029	5968.	30998.	0.	6300.	-0.	0.	6300.	217348.	5.0	2029
2030	5960.	29938.	0.	6120.	-0.	0.	6120.	223668.	5.0	2030
2031	5991.	29957.	0.	5973.	-0.	0.	5973.	229440.	5.0	2031
2032	6123.	30032.	0.	6047.	-0.	0.	6047.	235487.	5.0	2032
2033	5965.	30007.	0.	5990.	-0.	0.	5990.	241678.	5.0	2033
2034	6052.	30090.	0.	5968.	-0.	0.	5968.	247446.	5.0	2034
2035	6106.	30236.	0.	5960.	-0.	0.	5960.	253406.	5.0	2035
2036	6391.	30636.	0.	5991.	-0.	0.	5991.	259397.	5.0	2036
2037	6369.	30883.	0.	6123.	-0.	0.	6123.	265520.	5.0	2037
2038	6345.	31263.	0.	5965.	-0.	0.	5965.	271484.	5.0	2038
2039	6488.	31699.	0.	6052.	-0.	0.	6052.	277536.	5.0	2039
2040	6332.	31926.	0.	6106.	-0.	0.	6106.	283642.	5.0	2040
2041	0.	25535.	0.	6391.	-0.	0.	6391.	290033.	5.0	2041
2042	0.	19165.	0.	6369.	-0.	0.	6369.	296403.	5.0	2042
2043	0.	12620.	0.	6345.	-0.	0.	6345.	302748.	5.0	2043
2044	0.	6332.	0.	6488.	-0.	0.	6488.	309236.	5.0	2044
2045	0.	0.	0.	6332.	-0.	0.	6332.	315568.	5.0	2045
2046	0.	0.	0.	0.	-0.	0.	0.	315568.	0.0	2046
2047	0.	0.	0.	0.	-0.	0.	0.	315568.	0.0	2047
2048	0.	0.	0.	0.	-0.	0.	0.	315568.	0.0	2048
2049	0.	0.	0.	0.	-0.	0.	0.	315568.	0.0	2049
2050	0.	0.	0.	0.	-0.	0.	0.	315568.	0.0	2050
2051	0.	0.	0.	0.	-0.	0.	0.	315568.	0.0	2051
RAIL SHIPMENTS =			25672.4	46411.6		24538.9				
TRUCK SHIPMENTS =			26136.8	47251.1						

TABLE A.1.14. Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 4, MTU
2020 Repository

YEAR	REACTOR DISCHARGE	REACTOR STORAGE	SHIPMENT REACTOR TO AFR	SHIPMENT REACTOR TO REPOSITORY	AFR INVENTORY	SHIPMENT AFR TO REPOSITORY	REPOSITORY			YEAR
							RECEIPTS	INVENTORY	RECEIVING AGE, YEARS	
1980	1160.	7198.	0.	0.	0.	0.	0.	0.	0.0	1980
1981	1282.	8478.	0.	0.	0.	0.	0.	0.	0.0	1981
1982	1486.	9964.	0.	0.	0.	0.	0.	0.	0.0	1982
1983	1770.	10786.	949.	0.	949.	0.	0.	0.	0.0	1983
1984	2154.	11991.	949.	0.	1897.	0.	0.	0.	0.0	1984
1985	2335.	13271.	1056.	0.	2953.	0.	0.	0.	0.0	1985
1986	2605.	14666.	1209.	0.	4163.	0.	0.	0.	0.0	1986
1987	2830.	16336.	1160.	0.	5323.	0.	0.	0.	0.0	1987
1988	3045.	18099.	1282.	0.	6604.	0.	0.	0.	0.0	1988
1989	3159.	19772.	1486.	0.	8000.	0.	0.	0.	0.0	1989
1990	3368.	21370.	1770.	0.	9861.	0.	0.	0.	0.0	1990
1991	3648.	22862.	2154.	0.	12015.	0.	0.	0.	0.0	1991
1992	3858.	24384.	2335.	0.	14380.	0.	0.	0.	0.0	1992
1993	3938.	25717.	2605.	0.	16945.	0.	0.	0.	0.0	1993
1994	4235.	27121.	2830.	0.	19745.	0.	0.	0.	0.0	1994
1995	4380.	28457.	3045.	0.	22870.	0.	0.	0.	0.0	1995
1996	4588.	29886.	3159.	0.	25929.	0.	0.	0.	0.0	1996
1997	4854.	31371.	3368.	0.	29397.	0.	0.	0.	0.0	1997
1998	5083.	32804.	3648.	0.	33003.	0.	0.	0.	0.0	1998
1999	5303.	34254.	3858.	0.	36840.	0.	0.	0.	0.0	1999
2000	5599.	35915.	3938.	0.	40708.	0.	0.	0.	0.0	2000
2001	5854.	37534.	4235.	0.	45033.	0.	0.	0.	0.0	2001
2002	5757.	38911.	4380.	0.	49413.	0.	0.	0.	0.0	2002
2003	5761.	40083.	4588.	0.	54001.	0.	0.	0.	0.0	2003
2004	5705.	40934.	4854.	0.	58854.	0.	0.	0.	0.0	2004
2005	5779.	41631.	5083.	0.	63937.	0.	0.	0.	0.0	2005
2006	5639.	41967.	5303.	0.	69240.	0.	0.	0.	0.0	2006
2007	5740.	42107.	5599.	0.	74879.	0.	0.	0.	0.0	2007
2008	5661.	42107.	5661.	0.	80900.	0.	0.	0.	0.0	2008
2009	5763.	42166.	5705.	0.	86204.	0.	0.	0.	0.0	2009
2010	5850.	42237.	5779.	0.	91944.	0.	0.	0.	0.0	2010
2011	5761.	42339.	5639.	0.	97622.	0.	0.	0.	0.0	2011
2012	5883.	42504.	5740.	0.	103362.	0.	0.	0.	0.0	2012
2013	5999.	42839.	5661.	0.	109003.	0.	0.	0.	0.0	2013
2014	6263.	43338.	5763.	0.	114786.	0.	0.	0.	0.0	2014
2015	5868.	43452.	5754.	0.	120500.	0.	0.	0.	0.0	2015
2016	5844.	43535.	5761.	0.	126301.	0.	0.	0.	0.0	2016
2017	5797.	43532.	5799.	0.	132101.	0.	0.	0.	0.0	2017
2018	5618.	43532.	5618.	0.	137719.	0.	0.	0.	0.0	2018
2019	5520.	43532.	5520.	0.	143259.	0.	0.	0.	0.0	2019
2020	5810.	43532.	5110.	700.	148399.	0.	700.	700.	48.0	2020
2021	6064.	43532.	4764.	1300.	153112.	0.	1300.	2000.	46.4	2021
2022	6094.	43758.	4074.	1794.	156941.	206.	2000.	4000.	45.3	2022
2023	6309.	44224.	5507.	337.	160824.	1663.	2000.	6000.	44.4	2023
2024	6300.	44846.	5520.	157.	164502.	1843.	2000.	8000.	43.7	2024
2025	6120.	45228.	5182.	556.	167540.	2144.	2700.	10700.	42.9	2025
2026	5973.	45681.	4539.	981.	169760.	2319.	3300.	14000.	42.2	2026
2027	6047.	45918.	4965.	845.	171549.	3155.	4000.	18000.	41.6	2027
2028	5990.	45918.	5302.	688.	173560.	3312.	4000.	22000.	41.1	2028
2029	5968.	45918.	5442.	526.	175228.	3474.	4000.	26000.	40.7	2029
2030	5960.	45918.	5340.	620.	175688.	3580.	4000.	30000.	40.2	2030
2031	5991.	45918.	5024.	967.	173479.	7033.	4000.	40000.	39.2	2031
2032	6123.	45921.	5232.	885.	169549.	9112.	10000.	50000.	38.0	2032
2033	5965.	45921.	4884.	1081.	163563.	10919.	10000.	62000.	36.5	2033
2034	6052.	45926.	4776.	1271.	155611.	12729.	14000.	76000.	34.9	2034
2035	6106.	46041.	5187.	803.	147601.	13197.	14000.	90000.	33.4	2035
2036	6391.	46464.	5497.	71.	139569.	13929.	14000.	104000.	32.0	2036
2037	6369.	46874.	5960.	0.	131399.	14000.	14000.	118000.	30.5	2037
2038	6345.	47227.	5932.	59.	123520.	13941.	14000.	132000.	29.1	2038
2039	6488.	47593.	5992.	131.	115623.	13869.	14000.	146000.	27.7	2039
2040	6332.	47960.	4611.	1333.	107408.	12647.	14000.	140000.	26.3	2040
2041	0.	45127.	0.	2830.	96481.	11166.	14000.	174000.	25.0	2041
2042	0.	44597.	0.	730.	83171.	13270.	14000.	188000.	23.5	2042
2043	0.	41756.	0.	2641.	71012.	11359.	14000.	202000.	22.1	2043
2044	0.	39568.	0.	2188.	60000.	11812.	14000.	216000.	20.8	2044
2045	0.	37581.	0.	1987.	47947.	12013.	14000.	230000.	19.6	2045
2046	0.	35231.	0.	2330.	36347.	11650.	14000.	244000.	18.2	2046
2047	0.	33696.	0.	1533.	23070.	12467.	14000.	258000.	16.9	2047
2048	0.	33075.	0.	223.	10003.	13777.	14000.	272000.	15.6	2048
2049	0.	29568.	0.	3907.	0.	10093.	14000.	286000.	14.3	2049
2050	0.	15668.	0.	14000.	0.	0.	14000.	300000.	13.0	2050
2051	0.	1568.	0.	14000.	0.	0.	14000.	314000.	11.8	2051
2052	0.	0.	0.	1568.	0.	0.	1568.	315568.	11.6	2052
2053	0.	0.	0.	0.	0.	0.	0.	315568.	0.0	2053
2054	0.	0.	0.	0.	0.	0.	0.	315568.	0.0	2054
2055	0.	0.	0.	0.	0.	0.	0.	315568.	0.0	2055
2056	0.	0.	0.	0.	0.	0.	0.	315568.	0.0	2056
RAIL SHIPMENTS =			57673.1	14411.0		55126.6				
TRUCK SHIPMENTS =			58716.3	14671.6						

TABLE A.1.15. Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 5, MTU
2000 Repository

YEAR	REACTOR DISCHARGE	REACTOR STORAGE	SHIPMENT REACTOR TO AFR	SHIPMENT REACTOR TO REPOSITORY	AFR INVENTORY	SHIPMENT AFR TO REPOSITORY	REPOSITORY			YEAR
							RECEIPTS	INVENTORY	RECEIVING AGE, YEARS	
1980	1160.	7196.	0.	0.	0.	0.	0.	0.	0.0	1980
1981	1282.	8078.	0.	0.	0.	0.	0.	0.	0.0	1981
1982	1488.	9084.	0.	0.	0.	0.	0.	0.	0.0	1982
1983	1770.	10786.	909.	0.	909.	0.	0.	0.	0.0	1983
1984	2154.	11991.	909.	0.	1807.	0.	0.	0.	0.0	1984
1985	2335.	13271.	1056.	0.	2943.	0.	0.	0.	0.0	1985
1986	2605.	14666.	1209.	0.	4143.	0.	0.	0.	0.0	1986
1987	2830.	16336.	1160.	0.	5323.	0.	0.	0.	0.0	1987
1988	3045.	18099.	1282.	0.	6604.	0.	0.	0.	0.0	1988
1989	3159.	19772.	1486.	0.	8000.	0.	0.	0.	0.0	1989
1990	3368.	21370.	1770.	0.	9841.	0.	0.	0.	0.0	1990
1991	3646.	22862.	2154.	0.	12015.	0.	0.	0.	0.0	1991
1992	3858.	24384.	2335.	0.	14340.	0.	0.	0.	0.0	1992
1993	3938.	25717.	2605.	0.	16945.	0.	0.	0.	0.0	1993
1994	4235.	27121.	2830.	0.	19745.	0.	0.	0.	0.0	1994
1995	4380.	28457.	3045.	0.	22840.	0.	0.	0.	0.0	1995
1996	4588.	29886.	3159.	0.	25949.	0.	0.	0.	0.0	1996
1997	4854.	31371.	3368.	0.	29347.	0.	0.	0.	0.0	1997
1998	5083.	32808.	3646.	0.	33003.	0.	0.	0.	0.0	1998
1999	5303.	34254.	3858.	0.	36860.	0.	0.	0.	0.0	1999
2000	5599.	35615.	3938.	700.	40008.	0.	700.	700.	28.0	2000
2001	5854.	37034.	3935.	1300.	43043.	0.	1300.	2000.	26.4	2001
2002	5902.	39056.	2586.	1794.	45413.	206.	2000.	4000.	25.3	2002
2003	6051.	40580.	4251.	337.	48041.	1663.	2000.	6000.	24.4	2003
2004	6186.	41817.	4696.	157.	50848.	1843.	2000.	8000.	23.7	2004
2005	6376.	43105.	4597.	556.	53247.	2184.	2700.	10700.	22.9	2005
2006	6364.	44166.	4323.	981.	55220.	2319.	2300.	14000.	22.2	2006
2007	6626.	45192.	4754.	845.	56849.	3155.	2000.	18000.	21.6	2007
2008	6683.	46022.	5166.	688.	58603.	3312.	2000.	22000.	21.1	2008
2009	6906.	47026.	5377.	526.	60505.	3474.	2000.	26000.	20.7	2009
2010	7150.	48124.	5432.	620.	60647.	5380.	2000.	32000.	20.2	2010
2011	7225.	49204.	5179.	967.	58703.	7033.	2000.	40000.	19.2	2011
2012	7470.	50298.	4994.	1382.	55148.	8618.	10000.	50000.	18.0	2012
2013	7736.	51670.	1899.	4465.	49542.	7535.	12000.	62000.	16.5	2013
2014	8116.	53160.	4843.	1783.	44148.	10217.	12000.	74000.	15.1	2014
2015	7934.	54411.	3965.	2719.	38801.	9281.	12000.	86000.	13.9	2015
2016	7988.	55492.	5105.	1801.	33748.	10199.	12000.	94000.	12.9	2016
2017	8093.	56435.	5321.	1829.	28807.	10171.	12000.	110000.	12.0	2017
2018	8073.	57283.	3550.	3676.	24152.	8324.	12000.	122000.	11.1	2018
2019	8144.	57956.	1862.	5608.	19503.	6392.	12000.	134000.	10.3	2019
2020	8533.	58754.	3185.	4551.	15398.	7449.	12000.	146000.	9.6	2020
2021	8973.	59611.	4010.	4106.	11444.	7494.	12000.	158000.	9.0	2021
2022	9087.	60445.	0.	8053.	7407.	3947.	12000.	170000.	8.4	2022
2023	9500.	62276.	736.	7133.	3346.	4867.	12000.	182000.	7.9	2023
2024	9625.	62531.	0.	9370.	746.	2630.	12000.	194000.	7.4	2024
2025	9617.	60884.	0.	11264.	0.	736.	12000.	206000.	6.9	2025
2026	9559.	58443.	0.	12000.	0.	0.	12000.	218000.	6.4	2026
2027	9805.	56248.	0.	12000.	0.	0.	12000.	230000.	6.1	2027
2028	9931.	54180.	0.	12000.	0.	0.	12000.	242000.	5.7	2028
2029	10012.	52192.	0.	12000.	0.	0.	12000.	254000.	5.5	2029
2030	10141.	50333.	0.	12000.	0.	0.	12000.	266000.	5.2	2030
2031	10318.	50208.	0.	10443.	0.	0.	10443.	276443.	5.0	2031
2032	10593.	50995.	0.	9805.	0.	0.	9805.	286248.	5.0	2032
2033	10550.	51614.	0.	9931.	0.	0.	9931.	296180.	5.0	2033
2034	10819.	52017.	0.	10012.	0.	0.	10012.	306192.	5.0	2034
2035	10891.	53166.	0.	10141.	0.	0.	10141.	316333.	5.0	2035
2036	11334.	54182.	0.	10318.	0.	0.	10318.	326651.	5.0	2036
2037	11463.	55053.	0.	10593.	0.	0.	10593.	337244.	5.0	2037
2038	11522.	56025.	0.	10550.	0.	0.	10550.	347793.	5.0	2038
2039	11754.	56663.	0.	10819.	0.	0.	10819.	358600.	5.0	2039
2040	11815.	57887.	0.	10891.	0.	0.	10891.	369499.	5.0	2040
2041	0.	48554.	0.	11334.	0.	0.	11334.	380833.	5.0	2041
2042	0.	35090.	0.	11463.	0.	0.	11463.	392296.	5.0	2042
2043	0.	23569.	0.	11522.	0.	0.	11522.	403818.	5.0	2043
2044	0.	11815.	0.	11754.	0.	0.	11754.	415572.	5.0	2044
2045	0.	0.	0.	11815.	0.	0.	11815.	427387.	5.0	2045
2046	0.	0.	0.	0.	0.	0.	0.	427387.	0.0	2046
2047	0.	0.	0.	0.	0.	0.	0.	427387.	0.0	2047
2048	0.	0.	0.	0.	0.	0.	0.	427387.	0.0	2048
2049	0.	0.	0.	0.	0.	0.	0.	427387.	0.0	2049
2050	0.	0.	0.	0.	0.	0.	0.	427387.	0.0	2050
2051	0.	0.	0.	0.	0.	0.	0.	427387.	0.0	2051
RAIL SHIPMENTS *			29418.9	68207.5		28120.0				
TRUCK SHIPMENTS *			29951.0	69441.2						

TABLE A.1.16. Spent Fuel Logistics for the Once-Through Fuel Cycle--Growth Case 5, MTU

2020 Repository

YEAR	REACTOR DISCHARGE	REACTOR STORAGE	SHIPMENT REACTOR TO AFR	SHIPMENT REACTOR TO REPOSITORY	AFR INVENTORY	SHIPMENT AFR TO REPOSITORY	REPOSITORY			YEAR
							RECEIPTS	INVENTORY	RECEIVING AGE, YEARS	
1980	1160.	7196.	0.	0.	0.	0.	0.	0.	0.0	1980
1981	1282.	8078.	0.	0.	0.	0.	0.	0.	0.0	1981
1982	1486.	9064.	0.	0.	0.	0.	0.	0.	0.0	1982
1983	1770.	10786.	909.	0.	909.	0.	0.	0.	0.0	1983
1984	2154.	11991.	909.	0.	1807.	0.	0.	0.	0.0	1984
1985	2335.	13271.	1056.	0.	2963.	0.	0.	0.	0.0	1985
1986	2605.	14666.	1209.	0.	4143.	0.	0.	0.	0.0	1986
1987	2830.	16336.	1160.	0.	5353.	0.	0.	0.	0.0	1987
1988	3045.	18099.	1282.	0.	6664.	0.	0.	0.	0.0	1988
1989	3159.	19772.	1486.	0.	8000.	0.	0.	0.	0.0	1989
1990	3368.	21370.	1770.	0.	9841.	0.	0.	0.	0.0	1990
1991	3646.	22867.	2154.	0.	12015.	0.	0.	0.	0.0	1991
1992	3858.	24384.	2335.	0.	14360.	0.	0.	0.	0.0	1992
1993	3938.	25717.	2605.	0.	16949.	0.	0.	0.	0.0	1993
1994	4235.	27121.	2830.	0.	19745.	0.	0.	0.	0.0	1994
1995	4380.	28437.	3045.	0.	22840.	0.	0.	0.	0.0	1995
1996	4588.	29886.	3159.	0.	25949.	0.	0.	0.	0.0	1996
1997	4854.	31371.	3368.	0.	29347.	0.	0.	0.	0.0	1997
1998	5083.	32808.	3646.	0.	33063.	0.	0.	0.	0.0	1998
1999	5303.	34254.	3858.	0.	36840.	0.	0.	0.	0.0	1999
2000	5599.	35915.	3938.	0.	40708.	0.	0.	0.	0.0	2000
2001	5854.	37534.	4235.	0.	45023.	0.	0.	0.	0.0	2001
2002	5902.	39056.	4380.	0.	49413.	0.	0.	0.	0.0	2002
2003	6051.	40520.	4588.	0.	54001.	0.	0.	0.	0.0	2003
2004	6146.	41812.	4854.	0.	58844.	0.	0.	0.	0.0	2004
2005	6376.	43105.	5083.	0.	63947.	0.	0.	0.	0.0	2005
2006	6364.	44166.	5303.	0.	69200.	0.	0.	0.	0.0	2006
2007	6626.	45192.	5599.	0.	74849.	0.	0.	0.	0.0	2007
2008	6683.	46027.	5854.	0.	80603.	0.	0.	0.	0.0	2008
2009	6906.	47026.	5902.	0.	86505.	0.	0.	0.	0.0	2009
2010	7150.	48124.	6051.	0.	92647.	0.	0.	0.	0.0	2010
2011	7225.	49204.	6146.	0.	98703.	0.	0.	0.	0.0	2011
2012	7470.	50298.	6376.	0.	105144.	0.	0.	0.	0.0	2012
2013	7736.	51670.	6364.	0.	111542.	0.	0.	0.	0.0	2013
2014	8116.	53160.	6626.	0.	118146.	0.	0.	0.	0.0	2014
2015	7934.	54411.	6683.	0.	124801.	0.	0.	0.	0.0	2015
2016	7988.	55497.	6906.	0.	131749.	0.	0.	0.	0.0	2016
2017	8093.	56435.	7150.	0.	138807.	0.	0.	0.	0.0	2017
2018	8073.	57283.	7225.	0.	146192.	0.	0.	0.	0.0	2018
2019	8144.	57956.	7470.	0.	153503.	0.	0.	0.	0.0	2019
2020	8533.	58754.	7036.	700.	160628.	0.	700.	700.	48.0	2020
2021	8973.	59611.	6816.	1300.	167404.	0.	1300.	2000.	46.4	2021
2022	9087.	60764.	6140.	1798.	173375.	206.	2000.	4000.	45.3	2022
2023	9500.	62276.	7651.	337.	179346.	1663.	3000.	6000.	44.4	2023
2024	9625.	63808.	7935.	157.	185448.	1943.	3000.	8000.	43.7	2024
2025	9617.	65353.	7517.	556.	190841.	2144.	3700.	10700.	42.9	2025
2026	9559.	66768.	7163.	981.	195675.	2319.	3300.	14000.	42.2	2026
2027	9805.	68040.	7688.	845.	200208.	3155.	4000.	18000.	41.6	2027
2028	9931.	68999.	8285.	688.	205141.	3312.	4000.	22000.	41.1	2028
2029	10012.	69924.	8562.	526.	210248.	3474.	4000.	26000.	40.7	2029
2030	10141.	70565.	8880.	620.	213746.	3580.	4000.	32000.	40.2	2030

TABLE A.1.16. (Contd)

YEAR	REACTOR DISCHARGE	REACTOR STORAGE	SHIPMENT REACTOR TO AFR	SHIPMENT REACTOR TO REPOSITORY	AFR INVENTORY	SHIPMENT AFR TO REPOSITORY	REPOSITORY			YEAR
							RECEIPTS	INVENTORY	RECEIVING AGE, YEARS	
2031	10318.	71258.	8657.	967.	215393.	7033.	4000.	40000.	39.2	2031
2032	10593.	72234.	8730.	888.	215010.	9112.	14000.	50000.	38.0	2032
2033	10550.	73224.	8479.	1081.	212569.	10919.	12000.	62000.	36.5	2033
2034	10815.	74234.	8534.	1271.	208375.	12729.	18000.	76000.	34.9	2034
2035	10891.	75193.	8961.	970.	204386.	13030.	14000.	90000.	33.4	2035
2036	11334.	76515.	9452.	561.	200318.	13439.	12000.	104000.	32.1	2036
2037	11463.	77837.	9595.	546.	196459.	13454.	12000.	118000.	30.9	2037
2038	11522.	79041.	9707.	611.	192777.	13389.	14000.	132000.	29.8	2038
2039	11754.	80202.	9830.	763.	189370.	13237.	14000.	146000.	28.7	2039
2040	11815.	81467.	9842.	708.	185920.	13292.	14000.	160000.	27.8	2040
2041	0.	80307.	0.	1161.	173040.	12839.	14000.	174000.	27.0	2041
2042	0.	78356.	0.	1950.	161070.	12050.	14000.	188000.	26.3	2042
2043	0.	76239.	0.	2118.	149108.	11882.	14000.	202000.	25.5	2043
2044	0.	75426.	0.	812.	135941.	13188.	14000.	216000.	24.8	2044
2045	0.	72985.	0.	2441.	124422.	11559.	12000.	230000.	24.2	2045
2046	0.	70982.	0.	2003.	112465.	11997.	14000.	244000.	23.6	2046
2047	0.	69397.	0.	1585.	99990.	12415.	14000.	258000.	23.2	2047
2048	0.	68233.	0.	1164.	87153.	12836.	14000.	272000.	22.7	2048
2049	0.	66766.	0.	1468.	74621.	12532.	14000.	286000.	22.2	2049
2050	0.	64722.	0.	2044.	62645.	11956.	14000.	300000.	21.8	2050
2051	0.	62900.	0.	1822.	50447.	12178.	14000.	314000.	21.4	2051
2052	0.	61635.	0.	1265.	37742.	12735.	14000.	328000.	21.0	2052
2053	0.	60384.	0.	1250.	25022.	12750.	14000.	342000.	20.7	2053
2054	0.	59174.	0.	1210.	12213.	12790.	14000.	356000.	20.4	2054
2055	0.	57387.	0.	1787.	0.	12213.	14000.	370000.	20.1	2055
2056	0.	43387.	0.	14000.	0.	0.	14000.	384000.	19.8	2056
2057	0.	29387.	0.	14000.	0.	0.	14000.	398000.	19.6	2057
2058	0.	15387.	0.	14000.	0.	0.	14000.	412000.	19.4	2058
2059	0.	1387.	0.	14000.	0.	0.	14000.	426000.	19.2	2059
2060	0.	0.	0.	1387.	0.	0.	1387.	440000.	19.0	2060
2061	0.	0.	0.	0.	0.	0.	0.	427387.	19.6	2061
2062	0.	0.	0.	0.	0.	0.	0.	427387.	0.0	2062
2063	0.	0.	0.	0.	0.	0.	0.	427387.	0.0	2063
2064	0.	0.	0.	0.	0.	0.	0.	427387.	0.0	2064
2065	0.	0.	0.	0.	0.	0.	0.	427387.	0.0	2065
2066	0.	0.	0.	0.	0.	0.	0.	427387.	0.0	2066
RAIL SHIPMENTS *			75164.0	22462.4		71845.3				
TRUCK SHIPMENTS *			76523.6	22868.7						

TABLE A.1.17. Spent Fuel Logistics for the Reprocessing Fuel Cycle--Growth Case 3, MTU

1990 Reprocessing

YEAR	REACTOR DISCHARGE	REACTOR STORAGE	SHIPMENT REACTOR TO AFR	SHIPMENT REACTOR TO REPROCESS	AFR INVENTORY	SHIPMENT AFR TO REPROCESS	REPROCESS		RECEIVING FUEL, YEARS	YEAR
							ANNUAL	CUMULATIVE		
1980	1160.	7196.	0.	0.	0.	0.	0.	0.	0.0	1980
1981	1282.	8478.	0.	0.	0.	0.	0.	0.	0.0	1981
1982	1486.	9864.	0.	0.	0.	0.	0.	0.	0.0	1982
1983	1770.	10786.	949.	0.	949.	0.	0.	0.	0.0	1983
1984	2154.	11991.	949.	0.	1897.	0.	0.	0.	0.0	1984
1985	2335.	13271.	1056.	0.	2953.	0.	0.	0.	0.0	1985
1986	2605.	14666.	1209.	0.	4163.	0.	0.	0.	0.0	1986
1987	2830.	16336.	1160.	0.	5323.	0.	0.	0.	0.0	1987
1988	3043.	18099.	1282.	0.	6604.	0.	0.	0.	0.0	1988
1989	3159.	19772.	1486.	0.	8090.	0.	0.	0.	0.0	1989
1990	3368.	21370.	1070.	700.	9161.	0.	700.	700.	18.0	1990
1991	3646.	22867.	854.	1300.	10015.	0.	1300.	2000.	16.4	1991
1992	3858.	24384.	335.	2000.	10350.	0.	2000.	4000.	13.3	1992
1993	3938.	25717.	605.	2000.	10955.	0.	2000.	6000.	14.4	1993
1994	4235.	27121.	830.	2000.	11785.	0.	2000.	8000.	13.7	1994
1995	4380.	28457.	1688.	1357.	12170.	1343.	2700.	10700.	12.9	1995
1996	4588.	29886.	1527.	1632.	11999.	1668.	2300.	14000.	12.2	1996
1997	4854.	31371.	1311.	2097.	11367.	1943.	4000.	18000.	11.6	1997
1998	5083.	32808.	1463.	2183.	11003.	1817.	4000.	22000.	11.1	1998
1999	5303.	34254.	1616.	2241.	10840.	1759.	4000.	26000.	10.7	1999
2000	5399.	35915.	588.	3349.	10068.	1351.	4700.	30700.	10.4	2000
2001	5554.	37114.	0.	4654.	9442.	646.	4300.	36000.	9.9	2001
2002	5752.	37984.	0.	4883.	8385.	1117.	4000.	42000.	9.4	2002
2003	5749.	39488.	0.	4243.	5840.	2455.	4700.	48700.	8.9	2003
2004	5692.	40130.	0.	5050.	3670.	2250.	7300.	56000.	8.3	2004
2005	5766.	40362.	0.	5534.	1144.	2466.	4000.	64000.	7.7	2005
2006	5627.	38457.	0.	7536.	0.	1164.	4700.	72700.	7.0	2006
2007	5727.	34879.	0.	9300.	0.	0.	4300.	82000.	6.4	2007
2008	5647.	30526.	0.	10000.	0.	0.	10000.	92000.	5.7	2008
2009	5725.	25531.	0.	10700.	0.	0.	10700.	102700.	4.9	2009
2010	5788.	20039.	0.	11300.	0.	0.	11300.	114000.	4.0	2010
2011	5631.	13670.	0.	12000.	0.	0.	12000.	126000.	2.9	2011
2012	5700.	8550.	0.	10821.	0.	0.	10821.	136821.	2.0	2012
2013	5669.	8513.	0.	5706.	0.	0.	5706.	142527.	1.5	2013
2014	5796.	8652.	0.	5657.	0.	0.	5657.	148183.	1.5	2014
2015	5182.	7990.	0.	5844.	0.	0.	5844.	154027.	1.5	2015
2016	4993.	7510.	0.	5474.	0.	0.	5474.	159501.	1.5	2016
2017	4856.	7281.	0.	5084.	0.	0.	4084.	164585.	1.5	2017
2018	4370.	7014.	0.	4837.	0.	0.	4837.	169422.	1.5	2018
2019	4313.	6404.	0.	4824.	0.	0.	4824.	174246.	1.5	2019
2020	4653.	6832.	0.	4327.	0.	0.	4327.	178573.	1.5	2020
2021	4786.	7172.	0.	4447.	0.	0.	4447.	183020.	1.5	2021
2022	4617.	7001.	0.	4787.	0.	0.	4787.	187807.	1.5	2022
2023	4543.	6878.	0.	4666.	0.	0.	4666.	192473.	1.5	2023
2024	4150.	6368.	0.	4660.	0.	0.	4660.	197134.	1.5	2024
2025	3789.	5877.	0.	4280.	0.	0.	4280.	201814.	1.5	2025
2026	3375.	5220.	0.	4032.	0.	0.	4032.	205846.	1.5	2026
2027	3224.	4884.	0.	3560.	0.	0.	3560.	209006.	1.5	2027
2028	2948.	4524.	0.	3308.	0.	0.	3308.	212314.	1.5	2028
2029	2813.	4271.	0.	3066.	0.	0.	3066.	215379.	1.5	2029
2030	2598.	3932.	0.	2938.	0.	0.	2938.	218317.	1.5	2030

TABLE A.1.17. (Contd)

YEAR	REACTOR DISCHARGE	REACTOR STORAGE	SHIPMENT REACTOR TO AFR	SHIPMENT REACTOR TO REPROCESS	AFR INVENTORY	SHIPMENT AFR TO REPROCESS	REPROCESS			YEAR
							ANNUAL	CUMULATIVE	RECEIVING FUEL, YEARS	
1031	2353.	3606.	0.	2678.	0.	0.	2678.	220995.	1.5	2031
1032	2268.	3630.	0.	2445.	0.	0.	2845.	223440.	1.5	2032
1033	2033.	3126.	0.	2337.	0.	0.	2337.	225777.	1.5	2033
1034	1818.	2859.	0.	2083.	0.	0.	2083.	227862.	1.5	2034
1035	1732.	2403.	0.	1987.	0.	0.	1987.	229849.	1.5	2035
1036	1810.	2498.	0.	1715.	0.	0.	1987.	231564.	1.5	2036
1037	1519.	2414.	0.	1803.	0.	0.	1803.	233367.	1.5	2037
1038	1265.	1977.	0.	1702.	0.	0.	1702.	235060.	1.5	2038
1039	1194.	1797.	0.	1374.	0.	0.	1374.	236443.	1.5	2039
1040	752.	1317.	0.	1232.	0.	0.	1232.	237675.	1.5	2040
1041	0.	237.	0.	1040.	0.	0.	1040.	238755.	1.5	2041
1042	0.	0.	0.	237.	0.	0.	237.	238992.	1.7	2042
1043	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2043
1044	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2044
1045	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2045
1046	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2046
1047	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2047
1048	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2048
1049	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2049
1050	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2050
1051	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2051
1052	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2052
1053	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2053
1054	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2054
1055	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2055
1056	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2056
1057	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2057
1058	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2058
1059	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2059
1060	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2060
1061	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2061
1062	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2062
1063	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2063
1064	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2064
1065	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2065
1066	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2066
1067	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2067
1068	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2068
1069	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2069
1070	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2070
1071	0.	0.	0.	0.	0.	0.	0.	238992.	0.0	2071
TOTAL SHIPMENTS ■			4563.6	50028.9		4362.1				
TRUCK SHIPMENTS ■			4646.1	50933.4						

TABLE A.1.18. Spent Fuel Logistics for the Reprocessing Fuel Cycle--Growth Case 3, MTU

2010 Reprocessing

YEAR	REACTOR DISCHARGE	REACTOR STORAGE	SHIPMENT REACTOR TO AFR	SHIPMENT REACTOR TO REPROCESS	AFR INVENTORY	SHIPMENT AFR TO REPROCESS	REPROCESS			YEAR
							ANNUAL	CUMULATIVE	RECEIVING FUEL, YEARS	
1980	1160.	7196.	0.	0.	0.	0.	0.	0.	0.0	1980
1981	1282.	8078.	0.	0.	0.	0.	0.	0.	0.0	1981
1982	1486.	9064.	0.	0.	0.	0.	0.	0.	0.0	1982
1983	1770.	10786.	949.	0.	949.	0.	0.	0.	0.0	1983
1984	2154.	11991.	949.	0.	1897.	0.	0.	0.	0.0	1984
1985	2335.	13271.	1056.	0.	2953.	0.	0.	0.	0.0	1985
1986	2605.	14666.	1209.	0.	4163.	0.	0.	0.	0.0	1986
1987	2830.	16334.	1160.	0.	5323.	0.	0.	0.	0.0	1987
1988	3045.	18099.	1282.	0.	6604.	0.	0.	0.	0.0	1988
1989	3159.	19772.	1486.	0.	8000.	0.	0.	0.	0.0	1989
1990	3368.	21370.	1770.	0.	9841.	0.	0.	0.	0.0	1990
1991	3646.	22862.	2154.	0.	12015.	0.	0.	0.	0.0	1991
1992	3858.	24284.	2335.	0.	14300.	0.	0.	0.	0.0	1992
1993	3938.	25717.	2605.	0.	16905.	0.	0.	0.	0.0	1993
1994	4235.	27121.	2830.	0.	19785.	0.	0.	0.	0.0	1994
1995	4380.	28457.	3045.	0.	22840.	0.	0.	0.	0.0	1995
1996	4588.	29884.	3159.	0.	25909.	0.	0.	0.	0.0	1996
1997	4854.	31371.	3368.	0.	29347.	0.	0.	0.	0.0	1997
1998	5083.	32804.	3646.	0.	33003.	0.	0.	0.	0.0	1998
1999	5303.	34254.	3858.	0.	36840.	0.	0.	0.	0.0	1999
2000	5599.	35815.	3938.	0.	40768.	0.	0.	0.	0.0	2000
2001	5854.	37534.	4235.	0.	45023.	0.	0.	0.	0.0	2001
2002	5752.	38906.	4380.	0.	49413.	0.	0.	0.	0.0	2002
2003	5749.	40067.	4588.	0.	54001.	0.	0.	0.	0.0	2003
2004	5692.	40905.	4854.	0.	58860.	0.	0.	0.	0.0	2004
2005	5766.	41589.	5083.	0.	63927.	0.	0.	0.	0.0	2005
2006	5627.	41912.	5303.	0.	69200.	0.	0.	0.	0.0	2006
2007	5727.	42040.	5599.	0.	74849.	0.	0.	0.	0.0	2007
2008	5647.	42040.	5647.	0.	80486.	0.	0.	0.	0.0	2008
2009	5725.	42118.	5647.	0.	86123.	0.	0.	0.	0.0	2009
2010	5788.	42079.	5127.	700.	91240.	0.	700.	700.	38.0	2010
2011	5631.	42079.	4331.	1300.	95501.	0.	1300.	2000.	36.4	2011
2012	5700.	42079.	3700.	2000.	99201.	0.	2000.	4000.	35.3	2012
2013	5669.	42122.	3627.	2000.	102928.	0.	2000.	6000.	34.4	2013
2014	5796.	42191.	3727.	2000.	106644.	0.	2000.	8000.	33.7	2014
2015	5182.	42191.	3825.	1357.	109127.	1343.	2700.	10700.	32.9	2015
2016	4993.	42191.	3361.	1632.	110890.	1668.	3300.	14000.	32.2	2016
2017	4856.	42191.	2799.	2057.	111676.	1943.	4000.	18000.	31.6	2017
2018	4570.	42191.	2387.	2183.	112246.	1817.	4000.	22000.	31.1	2018
2019	4313.	42191.	2359.	1954.	112509.	2046.	4000.	26000.	30.7	2019
2020	4655.	42191.	2814.	1841.	112514.	2859.	4700.	30700.	30.4	2020
2021	4786.	42191.	2919.	1867.	112000.	3433.	5300.	36000.	29.9	2021
2022	4617.	42191.	2810.	1806.	110617.	4194.	6000.	42000.	29.4	2022
2023	4543.	42191.	2664.	1879.	108440.	4821.	6700.	48700.	28.9	2023
2024	4150.	42191.	2178.	1972.	105310.	5328.	7300.	56000.	28.3	2024
2025	3789.	42191.	1824.	1965.	101049.	6035.	8000.	64000.	27.7	2025
2026	3375.	42191.	1186.	2189.	95774.	6411.	8700.	72700.	27.0	2026
2027	3224.	42191.	915.	2309.	89668.	6991.	9300.	82000.	26.4	2027
2028	2948.	42191.	1068.	1880.	82677.	8120.	10000.	92000.	25.7	2028
2029	2813.	42191.	1646.	1167.	74760.	9533.	10700.	102700.	24.9	2029
2030	2598.	42191.	2304.	294.	66068.	11006.	11300.	114000.	24.0	2030

TABLE A.1.18. (Contd)

YEAR	REACTOR DISCHARGE	REACTOR STORAGE	SHIPMENT REACTOR TO AFB	SHIPMENT REACTOR TO REPROCESS	AFB INVENTORY	SHIPMENT AFB TO REPROCESS	REPROCESS		RECEIVING FUEL, YEARS	YEAR
							ANNUAL	CUMULATIVE		
2031	2353.	42191.	1950.	403.	96410.	11597.	12000.	126000.	22.9	2031
2032	2268.	41409.	0.	3051.	47441.	8949.	12000.	138000.	21.9	2032
2033	2033.	39209.	0.	4233.	39603.	7767.	12000.	150000.	20.7	2033
2034	1818.	36885.	0.	4142.	31875.	7458.	12000.	162000.	19.6	2034
2035	1732.	32915.	0.	5702.	25577.	6298.	12000.	174000.	18.3	2035
2036	1810.	30066.	0.	4660.	18106.	7340.	12000.	186000.	16.7	2036
2037	1519.	26240.	0.	3344.	11501.	6456.	12000.	198000.	15.1	2037
2038	1265.	19898.	0.	7607.	7108.	4393.	12000.	210000.	13.4	2038
2039	1194.	16053.	0.	5039.	187.	6961.	12000.	222000.	10.8	2039
2040	752.	6992.	0.	9813.	-0.	187.	12000.	232000.	7.4	2040
2041	0.	237.	0.	6755.	-0.	0.	6755.	238755.	3.8	2041
2042	0.	-0.	0.	237.	-0.	0.	237.	23992.	1.7	2042
2043	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2043
2044	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2044
2045	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2045
2046	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2046
2047	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2047
2048	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2048
2049	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2049
2050	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2050
2051	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2051
2052	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2052
2053	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2053
2054	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2054
2055	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2055
2056	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2056
2057	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2057
2058	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2058
2059	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2059
2060	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2060
2061	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2061
2062	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2062
2063	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2063
2064	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2064
2065	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2065
2066	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2066
2067	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2067
2068	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2068
2069	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2069
2070	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2070
2071	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2071
2072	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2072
2073	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2073
2074	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2074
2075	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2075
2076	0.	-0.	0.	0.	-0.	0.	0.	23992.	0.0	2076
RAIL SHIPMENTS ■			33271.3	21320.8		31902.7				
TRUCK SHIPMENTS ■			33873.1	21706.4						

TABLE A.1.19. Spent Fuel Logistics for the Reprocessing Fuel Cycle--Growth Case 4, MTU

2000 Reprocessing

YEAR	REACTOR DISCHARGE	REACTOR STORAGE	SHIPMENT REACTOR TO AFF	SHIPMENT REACTOR TO REPROCESS	AFF INVENTORY	SHIPMENT AFF TO REPROCESS	REPROCESS		RECEIVING FUEL, YEARS	YEAR
							ANNUAL	CUMULATIVE		
1980	1160.	7196.	0.	0.	0.	0.	0.	0.	0.0	1980
1981	1282.	8478.	0.	0.	0.	0.	0.	0.	0.0	1981
1982	1486.	9964.	0.	0.	0.	0.	0.	0.	0.0	1982
1983	1770.	10786.	949.	0.	949.	0.	0.	0.	0.0	1983
1984	2154.	11931.	949.	0.	1897.	0.	0.	0.	0.0	1984
1985	2335.	13271.	1056.	0.	2953.	0.	0.	0.	0.0	1985
1986	2605.	14666.	1209.	0.	4143.	0.	0.	0.	0.0	1986
1987	2830.	16336.	1160.	0.	5323.	0.	0.	0.	0.0	1987
1988	3045.	18099.	1282.	0.	6604.	0.	0.	0.	0.0	1988
1989	3159.	19772.	1486.	0.	8000.	0.	0.	0.	0.0	1989
1990	3368.	21370.	1770.	0.	9441.	0.	0.	0.	0.0	1990
1991	3646.	22867.	2154.	0.	12015.	0.	0.	0.	0.0	1991
1992	3858.	24384.	2335.	0.	14300.	0.	0.	0.	0.0	1992
1993	3938.	25717.	2605.	0.	16945.	0.	0.	0.	0.0	1993
1994	4235.	27121.	2830.	0.	19745.	0.	0.	0.	0.0	1994
1995	4380.	28457.	3045.	0.	22840.	0.	0.	0.	0.0	1995
1996	4588.	29886.	3159.	0.	25949.	0.	0.	0.	0.0	1996
1997	4854.	31371.	3368.	0.	29347.	0.	0.	0.	0.0	1997
1998	5083.	32808.	3646.	0.	33003.	0.	0.	0.	0.0	1998
1999	5303.	34254.	3858.	0.	36840.	0.	0.	0.	0.0	1999
2000	5599.	35915.	3238.	700.	40004.	0.	700.	700.	28.0	2000
2001	5854.	37534.	2935.	1300.	43033.	0.	1300.	2000.	26.4	2001
2002	5757.	38911.	2380.	2000.	45413.	0.	2000.	4000.	25.3	2002
2003	5761.	40083.	2588.	2000.	48001.	0.	2000.	6000.	24.4	2003
2004	5705.	40934.	2854.	2000.	50844.	0.	2000.	8000.	23.7	2004
2005	5779.	41631.	3726.	1357.	53247.	1343.	2700.	10700.	22.9	2005
2006	5639.	41967.	3671.	1632.	55207.	1662.	3300.	14000.	22.2	2006
2007	5740.	42107.	3542.	2057.	56849.	1943.	4000.	18000.	21.6	2007
2008	5661.	42107.	3478.	2183.	58500.	1817.	4000.	22000.	21.1	2008
2009	5763.	42114.	3803.	1954.	60247.	2046.	6000.	28000.	20.7	2009
2010	5850.	42203.	3919.	1841.	61317.	2859.	4700.	32700.	20.4	2010
2011	5761.	42260.	3837.	1867.	61722.	3433.	4300.	37000.	19.9	2011
2012	5885.	42366.	3973.	1806.	61501.	4194.	4000.	42000.	19.4	2012
2013	5995.	42722.	3760.	1879.	60400.	4821.	4700.	47700.	18.9	2013
2014	6263.	43245.	3767.	1972.	58800.	5328.	7300.	56000.	18.3	2014
2015	5868.	43452.	3696.	1965.	56500.	6035.	8000.	64000.	17.7	2015
2016	5844.	43532.	2506.	3257.	53604.	5443.	8700.	72700.	17.0	2016
2017	5797.	43532.	750.	5047.	50101.	4253.	9300.	82000.	16.4	2017
2018	5618.	43532.	258.	5360.	45719.	4640.	10000.	92000.	15.7	2018
2019	5520.	43532.	1427.	4093.	40549.	6607.	10700.	102700.	14.9	2019
2020	5810.	43532.	1566.	4244.	35049.	7056.	11300.	114000.	14.0	2020
2021	6064.	43532.	1959.	4105.	29112.	7895.	12000.	126000.	13.0	2021
2022	6094.	43758.	1924.	3944.	22981.	8056.	12000.	138000.	11.9	2022
2023	6309.	44224.	1568.	4276.	16824.	7724.	12000.	150000.	10.8	2023
2024	6300.	44727.	318.	5479.	10621.	6521.	12000.	162000.	9.8	2024
2025	6120.	40633.	0.	10213.	8845.	1787.	12000.	178000.	8.8	2025
2026	5973.	37452.	0.	9153.	5988.	2847.	12000.	186000.	7.7	2026
2027	6047.	35397.	0.	8103.	2000.	3897.	12000.	198000.	6.6	2027
2028	5990.	31825.	0.	9962.	43.	2038.	12000.	210000.	5.6	2028
2029	5968.	25446.	0.	11947.	-0.	53.	12000.	222000.	4.7	2029
2030	5960.	21406.	0.	10000.	-0.	0.	10000.	232000.	3.9	2030

TABLE A.1.19. (Contd)

YEAR	REACTOR DISCHARGE	REACTOR STORAGE	SHIPMENT REACTOR TO AFR	SHIPMENT REACTOR TO REPROCESS	AFR INVENTORY	SHIPMENT AFR TO REPROCESS	REPROCESS		RECEIVING FUEL, YEARS	YEAR
							ANNUAL	CUMULATIVE		
2031	5991.	17397.	0.	10000.	-0.	0.	10000.	242000.	3.2	2031
2032	6123.	13920.	0.	10000.	-0.	0.	10000.	252000.	2.6	2032
2033	5965.	9484.	0.	10000.	-0.	0.	10000.	262000.	1.9	2033
2034	6052.	9059.	0.	6477.	-0.	0.	6477.	268477.	1.5	2034
2035	6106.	9094.	0.	6071.	-0.	0.	6071.	274548.	1.5	2035
2036	6391.	9507.	0.	5978.	-0.	0.	5978.	280526.	1.5	2036
2037	6369.	9604.	0.	6272.	-0.	0.	6272.	286798.	1.5	2037
2038	6345.	9501.	0.	6488.	-0.	0.	6488.	293287.	1.5	2038
2039	6488.	9699.	0.	6290.	-0.	0.	6290.	299877.	1.5	2039
2040	6332.	9610.	0.	6421.	-0.	0.	6421.	306598.	1.5	2040
2041	0.	3094.	0.	6516.	-0.	0.	6516.	312470.	1.5	2041
2042	0.	-0.	0.	3094.	-0.	0.	3094.	315564.	1.7	2042
2043	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2043
2044	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2044
2045	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2045
2046	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2046
2047	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2047
2048	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2048
2049	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2049
2050	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2050
2051	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2051
2052	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2052
2053	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2053
2054	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2054
2055	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2055
2056	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2056
2057	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2057
2058	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2058
2059	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2059
2060	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2060
2061	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2061
2062	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2062
2063	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2063
2064	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2064
2065	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2065
2066	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2066
2067	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2067
2068	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2068
2069	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2069
2070	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2070
2071	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2071
2072	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2072
2073	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2073
2074	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2074
2075	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2075
2076	0.	-0.	0.	0.	-0.	0.	0.	315564.	0.0	2076
RAIL SHIPMENTS			23825.6	48258.5		22773.6				
TRUCK SHIPMENTS			24296.5	49131.4						

TABLE A.1.20. Spent Fuel Logistics for the Reprocessing Fuel Cycle--Growth Case 5, MTU

2000 Reprocessing

YEAR	REACTOR DISCHARGE	REACTOR STORAGE	SHIPMENT TO AFR	SHIPMENT REACTOR TO REPROCESS	AFR INVENTORY	SHIPMENT AFR TO REPROCESS	REPROCESS			YEAR
							ANNUAL	CUMULATIVE	RECEIVING FUEL, YEARS	
1980	1160.	7196.	0.	0.	0.	0.	0.	0.	0.0	1980
1981	1282.	8478.	0.	0.	0.	0.	0.	0.0	0.0	1981
1982	1486.	9864.	0.	0.	0.	0.	0.	0.0	0.0	1982
1983	1770.	10786.	249.	0.	249.	0.	0.	0.0	0.0	1983
1984	2154.	11991.	949.	0.	1907.	0.	0.	0.0	0.0	1984
1985	2335.	13271.	1056.	0.	2963.	0.	0.	0.0	0.0	1985
1986	2605.	14866.	1209.	0.	4173.	0.	0.	0.0	0.0	1986
1987	2830.	16736.	1410.	0.	5583.	0.	0.	0.0	0.0	1987
1988	3045.	18099.	1682.	0.	7265.	0.	0.	0.0	0.0	1988
1989	3159.	19772.	1946.	0.	9211.	0.	0.	0.0	0.0	1989
1990	3368.	21370.	2270.	0.	11481.	0.	0.	0.0	0.0	1990
1991	3646.	22862.	2754.	0.	14235.	0.	0.	0.0	0.0	1991
1992	3858.	24384.	3335.	0.	17570.	0.	0.	0.0	0.0	1992
1993	3938.	25717.	3605.	0.	21175.	0.	0.	0.0	0.0	1993
1994	4235.	27121.	3830.	0.	25005.	0.	0.	0.0	0.0	1994
1995	4380.	28057.	3045.	0.	28050.	0.	0.	0.0	0.0	1995
1996	4588.	29886.	3159.	0.	31209.	0.	0.	0.0	0.0	1996
1997	4854.	31371.	3368.	0.	34577.	0.	0.	0.0	0.0	1997
1998	5083.	32808.	3646.	0.	38223.	0.	0.	0.0	0.0	1998
1999	5303.	34254.	3956.	0.	42179.	0.	0.	0.0	0.0	1999
2000	5599.	35915.	3238.	700.	40008.	0.	700.	700.	28.0	2000
2001	5854.	37534.	2935.	1300.	43043.	0.	1300.	2000.	26.4	2001
2002	5902.	39056.	2380.	2000.	45413.	0.	2000.	4000.	25.3	2002
2003	6051.	40520.	2588.	2000.	48001.	0.	2000.	6000.	24.4	2003
2004	6146.	41812.	2854.	2000.	50844.	0.	2000.	8000.	23.7	2004
2005	6376.	43105.	3726.	1357.	53247.	1343.	2700.	10700.	22.9	2005
2006	6364.	44166.	3671.	1632.	55200.	1668.	3300.	14000.	22.2	2006
2007	6626.	45192.	3542.	2057.	56849.	1943.	4000.	18000.	21.6	2007
2008	6683.	46022.	3671.	2183.	58603.	1917.	4000.	22000.	21.1	2008
2009	6906.	47026.	3949.	1954.	60505.	2046.	4000.	26000.	20.7	2009
2010	7150.	48124.	4210.	1841.	61907.	2859.	4700.	30700.	20.4	2010
2011	7225.	49204.	4279.	1867.	62703.	3033.	5300.	36000.	19.9	2011
2012	7470.	50298.	4570.	1806.	63182.	4194.	6000.	42000.	19.4	2012
2013	7736.	51670.	4485.	1879.	62842.	4821.	6700.	48700.	18.9	2013
2014	8116.	53160.	4654.	1972.	62148.	5328.	7300.	56000.	18.3	2014
2015	7934.	54411.	4718.	1965.	60801.	6035.	8000.	64000.	17.7	2015
2016	7988.	55492.	3650.	3257.	59048.	5443.	8700.	72700.	17.0	2016
2017	8093.	56439.	2103.	5047.	56807.	4254.	9300.	82000.	16.4	2017
2018	8073.	57283.	1709.	5517.	54122.	4463.	10000.	92000.	15.7	2018
2019	8144.	57956.	2855.	4615.	50803.	6085.	10700.	102700.	15.0	2019
2020	8533.	58754.	2807.	4929.	47328.	6371.	11300.	114000.	14.3	2020
2021	8973.	59611.	2983.	5132.	43444.	6868.	12000.	126000.	13.5	2021
2022	9087.	60764.	2906.	5029.	38678.	7672.	12700.	138700.	12.7	2022
2023	9500.	62276.	2900.	5087.	33366.	8213.	13300.	152000.	11.9	2023
2024	9625.	63808.	2454.	5639.	27468.	8361.	14000.	166000.	11.0	2024
2025	9617.	65353.	405.	7578.	20841.	7122.	14700.	180700.	10.2	2025
2026	9559.	63278.	0.	11634.	17145.	3666.	15300.	196000.	9.4	2026
2027	9805.	62467.	0.	10416.	11581.	5584.	16000.	212000.	8.4	2027
2028	9931.	61515.	0.	11083.	5944.	5617.	16700.	228700.	7.5	2028
2029	10012.	59272.	0.	12255.	919.	5045.	17300.	246000.	6.6	2029
2030	10141.	54333.	0.	15081.	0.	219.	18000.	264000.	5.8	2030

TABLE A.1.20. (Contd)

YEAR	REACTOR DISCHARGE	REACTOR STORAGE	SHIPMENT REACTOR TO APF	SHIPMENT REACTOR TO REPROCESS	APF INVENTORY	REPROCESS		RECEIVING FUEL, YEARS	YEAR	
						ANNUAL	CUMULATIVE			
2031	10318.	48651.	0.	14000.	-0.	0.	14000.	270000.	5.2	2031
2032	10593.	43244.	0.	14000.	-0.	0.	14000.	284000.	4.5	2032
2033	10550.	37793.	0.	14000.	-0.	0.	14000.	310000.	3.9	2033
2034	10819.	32609.	0.	14000.	-0.	0.	14000.	326000.	3.3	2034
2035	10891.	29499.	0.	14000.	-0.	0.	14000.	340000.	2.9	2035
2036	11334.	26833.	0.	14000.	-0.	0.	14000.	354000.	2.5	2036
2037	11463.	24296.	0.	14000.	-0.	0.	14000.	348000.	2.3	2037
2038	11522.	21818.	0.	14000.	-0.	0.	14000.	342000.	2.0	2038
2039	11750.	19572.	0.	14000.	-0.	0.	14000.	346000.	1.8	2039
2040	11815.	17758.	0.	13629.	-0.	0.	13629.	409620.	1.6	2040
2041	0.	5917.	0.	11841.	-0.	0.	11841.	421470.	1.5	2041
2042	0.	-0.	0.	5917.	-0.	0.	5917.	427387.	1.4	2042
2043	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2043
2044	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2044
2045	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2045
2046	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2046
2047	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2047
2048	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2048
2049	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2049
2050	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2050
2051	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2051
2052	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2052
2053	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2053
2054	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2054
2055	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2055
2056	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2056
2057	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2057
2058	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2058
2059	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2059
2060	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2060
2061	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2061
2062	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2062
2063	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2063
2064	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2064
2065	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2065
2066	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2066
2067	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2067
2068	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2068
2069	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2069
2070	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2070
2071	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2071
2072	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2072
2073	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2073
2074	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2074
2075	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2075
2076	0.	-0.	0.	0.	-0.	0.	0.	427387.	0.0	2076
RAIL SHIPMENTS			27682.9	69943.6		26460.4				
TRUCK SHIPMENTS			28183.6	71208.7						

TABLE A.1.21. Number of Containers Sent to Repository in Once-Through Cases

<u>Growth Assumption</u>	<u>Repository Date</u>	<u>BWR Canisters</u>	<u>PWR Canisters</u>	<u>Total Canisters</u>
Present Inventory	1990	2.37×10^4	1.19×10^4	3.56×10^4
	2010	2.37×10^4	1.19×10^4	3.56×10^4
	2030	2.37×10^4	1.19×10^4	3.56×10^4
Present Capacity to Retirement	1990	1.04×10^5	6.12×10^4	1.65×10^5
	2010	1.04×10^5	6.12×10^4	1.65×10^5
	2030	1.04×10^5	6.12×10^4	1.65×10^5
250 GWe in 2000 and Decline to 0 in 2040	1990	4.91×10^5	3.17×10^5	8.08×10^5
	2010	4.91×10^5	3.17×10^5	8.08×10^5
	2030	4.91×10^5	3.17×10^5	8.08×10^5
250 GWe in 2000 and Steady State to 2040	2000	6.46×10^5	4.20×10^5	1.07×10^6
	2020	6.46×10^5	4.20×10^5	1.07×10^6
250 GWe in 2000 and Increase to 500 GWe in 2040	2000	8.71×10^5	5.70×10^5	1.44×10^6
	2020	8.71×10^5	5.70×10^5	1.44×10^6

TABLE A.1.22. Number of Containers Sent to Repository in Reprocessing Cases

Growth Assumption	Reprocessing Date	Repository Date	HLW Canisters				RH-TRU Canisters			RH-TRU Drums			CH-TRU(a)	
			Salt 3.2 kW/Can	Granite 1.7 kW/Can	Shale 1.2 kW/Can	Basalt 1.3 kW/Can	10 + R/hr	1-10 R/hr	.2-1 R/hr	10 + R/hr	1-10 R/hr	.2-1 R/hr	Drums	Boxes
250 GWe in 2000 and Decline to 0 in 2040	1990	1990	1.28×10^5	2.37×10^5	4.29×10^5	3.74×10^5	5.71×10^4	4.78×10^2	8.01×10^3	5.06×10^5	2.35×10^5	2.31×10^5	7.81×10^5	1.14×10^4
		2010	1.02×10^5	1.81×10^5	2.64×10^5	2.28×10^5	5.71×10^4	4.78×10^2	8.01×10^3	5.06×10^5	2.35×10^5	2.31×10^5	7.81×10^5	1.14×10^4
	2010	2010	7.87×10^4	1.17×10^5	1.77×10^5	1.69×10^5	5.71×10^4	4.78×10^2	8.01×10^3	5.06×10^5	2.35×10^5	2.31×10^5	5.34×10^5	8.86×10^3
		2030	8.17×10^4	1.14×10^5	1.60×10^5	1.47×10^5	5.71×10^4	4.78×10^2	8.01×10^3	5.06×10^5	2.35×10^5	2.31×10^5	7.81×10^5	1.14×10^4
250 GWe in 2000 and Steady State to 2040	2000	2000	1.37×10^5	2.48×10^5	3.47×10^5	3.09×10^5	7.54×10^4	6.31×10^2	1.06×10^4	6.68×10^5	3.11×10^5	3.06×10^5	8.59×10^5	1.34×10^4
		2020	1.14×10^5	1.92×10^5	2.69×10^5	2.51×10^5	7.54×10^4	6.31×10^2	1.06×10^4	6.68×10^5	3.11×10^5	3.06×10^5	8.59×10^5	1.34×10^4
250 GWe in 2000 and Increase to 500 GWe in 2040	2000	2000	1.87×10^5	3.59×10^5	5.32×10^5	4.54×10^5	1.02×10^5	8.55×10^2	1.43×10^4	9.05×10^5	4.21×10^5	4.14×10^5	1.20×10^6	1.86×10^4
		2020	1.61×10^5	2.76×10^5	3.88×10^5	3.65×10^5	1.02×10^5	8.55×10^2	1.43×10^4	9.05×10^5	4.21×10^5	4.14×10^5	1.20×10^6	1.86×10^4

(a) Includes waste from FRP and MOX decommissioning.

TABLE A.1.23. Age of HLW at Time of Disposal

Year	Case: Reprocessing: Repository:	3	3	3	3	3	4	4	5	5
		1990	1990	2010	1990	2010	2000	2000	2000	2000
1995		23.0								
1996		21.4								
1997		20.3								
1998		19.4								
1999		18.7								
2000		17.9								
2001		17.2								
2002		16.6								
2003		16.1								
2004		15.7								
2005		15.4					33.0		33.0	
2006		14.9					31.4		31.4	
2007		14.4					30.3		30.3	
2008		13.9					29.4		29.4	
2009		13.3					28.7		28.7	
2010		12.7	38.0				27.9		27.9	
2011		12.0	36.4				27.2		27.2	
2012		11.4	35.3				26.6		26.6	
2013		10.7	34.4				26.1		26.1	
2014		9.9	33.7				25.7		25.7	
2015		9.0	32.9	43.0			25.4		25.4	
2016		7.9	32.2	41.4			24.9		24.9	
2017		7.0	31.6	40.3			24.4		24.4	
2018		6.5	31.1	39.4			23.9		23.9	
2019		6.5	30.7	38.7			23.3		23.3	
2020		6.5	29.7	37.9			22.7	48.0	22.7	48.0
2021		6.5	28.1	37.2			22.0	46.4	22.0	46.4
2022		6.5	26.4	36.6			21.4	45.3	21.4	45.3
2023		6.5	24.7	36.1			20.7	44.4	20.7	44.4
2024		6.5	22.6	35.7			19.9	43.7	20.0	43.7
2025		6.5	20.6	35.4			19.0	42.9	19.3	42.9
2026		6.5	18.3	34.9			18.0	42.2	18.5	42.2
2027		6.5	16.4	34.4			16.9	41.6	17.7	41.6
2028		6.5	15.1	33.9			15.8	41.1	16.9	41.1
2029		6.5	14.4	33.3			14.8	40.7	16.0	40.7
2030		6.5	13.7	32.7	58.0	58.0	13.8	39.7	15.2	39.7
2031		6.5	13.0	32.0	56.4	56.4	12.7	38.1	14.4	38.1
2032		6.5	12.0	31.4	55.3	55.3	11.6	36.4	13.4	36.4
2033		6.5	11.3	30.7	54.4	54.4	10.6	34.7	12.5	34.7
2034		6.5	10.7	29.9	53.7	53.7	9.7	32.6	11.6	32.7
2035		6.5	10.0	29.0	52.9	52.9	8.9	30.6	10.8	30.7
2036		6.5	9.1	27.9	52.2	52.2	8.2	28.2	10.2	28.7
2037		6.5	8.4	26.9	51.6	51.6	7.6	25.7	9.5	26.6
2038		6.5	7.7	25.7	51.1	51.1	6.9	23.0	8.9	24.3
2039		6.5	7.0	24.6	50.7	50.7	6.5	20.4	8.3	22.0
2040		6.5	6.5	23.3	48.9	49.7	6.5	17.9	7.9	19.7
2041		6.5	6.5	21.7	45.7	48.1	6.5	15.8	7.6	17.6
2042		6.5	6.5	20.1	42.3	46.4	6.5	13.5	7.3	15.5
2043		6.5	6.5	18.4	38.6	44.7	6.5	11.9	7.0	13.7
2044		6.5	6.5	15.8	34.6	42.6	6.5	10.7	6.8	12.2
2045		6.5	6.5	12.4	31.9	40.6	6.5	9.9	6.6	10.8
2046		6.5	6.5	8.8	30.2	38.2	6.5	9.1	6.5	9.5
2047		6.7	6.7	6.7	28.4	35.6	6.7	8.4	6.8	8.5
2048					26.6	32.6		8.1		8.1
2049					24.9	29.3				
2050					23.2	25.6				
2051					21.5	20.5				
2052					19.7	14.7				
2053					17.9	12.7				
2054					16.3					
2055					15.3					

A.2 RADIOACTIVE INVENTORY TABLES

The radioactivity inventory tables (A.2.1a through A.2.9b) differ from similar tables in Chapter 7 by showing the inventory of each major radionuclide as a function of time. These tables appear in sets of two tables; one table shows the inventory of fission and activation products and the other table shows the inventory of actinides. Tables are provided only for the different growth cases in the once-through cycle since repository opening dates have no effect on the inventory after the year 2070. Two tables are shown for the Reprocessing Case 3. This case has two different reprocessing startup dates and inventories are a function of reprocessing startup time, since that controls the amount of fuel that is recycled, which affects the quantities of plutonium and other actinides in the wastes.

TABLE A.2.1a. Radioactivity Inventory--Once-Through Cycle--Growth Case 1, Curies(A)

Fission and Activation Products

MAJOR RADIONUCLIDES	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
H-3	1.01E+06	6.07E+04	1.97E+04	2.31E+06	0.	0.	0.	0.	0.	0.	0.
C-14	6.17E+03	6.13E+03	6.11E+03	5.82E+03	5.82E+03	3.36E+03	1.85E+03	1.27E+01	3.88E+02	0.	0.
MN-54	1.55E+02	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
FE-55	9.18E+05	1.50E+00	7.30E+03	0.	0.	0.	0.	0.	0.	0.	0.
CO-60	3.86E+06	5.34E+03	3.83E+02	0.	0.	0.	0.	0.	0.	0.	0.
NI-64	2.32E+04	2.32E+04	2.32E+04	2.31E+04	2.40E+04	2.22E+04	2.13E+04	1.40E+04	9.76E+03	3.06E+02	4.03E+00
NI-63	3.18E+06	2.18E+06	1.88E+06	5.82E+04	2.04E+03	1.70E+10	0.	0.	0.	0.	0.
SE-79	2.63E+03	2.62E+03	2.62E+03	2.61E+03	2.60E+03	2.49E+03	2.46E+03	1.44E+03	9.95E+02	1.27E+01	6.18E-02
KR-85	2.17E+07	8.80E+05	2.04E+05	1.20E-06	0.	0.	0.	0.	0.	0.	0.
RA-227	1.31E+01	1.31E+01	1.31E+01	1.31E+01	1.41E+01	1.31E+01	1.41E+01	1.41E+01	1.31E+01	1.31E+01	1.31E+01
BR-90+Y-90	4.27E+04	1.83E+08	1.12E+08	5.03E+03	2.33E-02	0.	0.	0.	0.	0.	0.
ZR-93	1.25E+04	1.25E+04	1.25E+04	1.25E+04	1.25E+04	1.25E+04	1.25E+04	1.22E+04	1.19E+04	9.93E+03	7.88E+03
NB-93M	8.28E+03	1.22E+04	1.24E+04	1.25E+04	1.25E+04	1.25E+04	1.25E+04	1.22E+04	1.20E+04	9.94E+03	7.89E+03
TC-99	9.69E+04	9.69E+04	9.68E+04	9.67E+04	9.66E+04	9.53E+04	9.47E+04	8.21E+04	6.07E+04	1.86E+04	3.58E+03
RU-106+RH-106	9.02E+05	9.67E+10	0.	0.	0.	0.	0.	0.	0.	0.	0.
PD-107	7.05E+02	7.05E+02	7.05E+02	7.05E+02	7.05E+02	7.05E+02	7.05E+02	7.02E+02	6.98E+02	6.71E+02	6.39E+02
AG-110M	4.78E+01	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
CO-113M	1.89E+04	1.59E+04	5.92E+02	1.12E-06	0.	0.	0.	0.	0.	0.	0.
OB-125+TE-125M	1.10E+06	2.96E+00	1.75E-02	0.	0.	0.	0.	0.	0.	0.	0.
BN-126+PB-126	1.03E+04	1.03E+04	1.03E+04	1.03E+04	1.03E+04	9.99E+03	9.65E+03	7.41E+03	5.17E+03	3.24E+02	1.02E+01
I-129	2.45E+02	2.45E+02	2.45E+02	2.45E+02	2.45E+02	2.45E+02	2.45E+02	2.45E+02	2.44E+02	2.40E+02	2.35E+02
CB-134	4.51E+06	2.53E+01	2.94E-04	0.	0.	0.	0.	0.	0.	0.	0.
CB-135	1.92E+03	1.92E+03	1.92E+03	1.92E+03	1.92E+03	1.92E+03	1.91E+03	1.90E+03	1.88E+03	1.71E+03	1.52E+03
CB-137+BA-137	8.59E+08	2.71E+08	1.71E+08	1.45E+04	1.40E+01	0.	0.	0.	0.	0.	0.
CE-144+PR-144	3.60E+05	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
PM-147	1.78E+07	3.73E+01	1.63E-01	0.	0.	0.	0.	0.	0.	0.	0.
SM-151	4.05E+06	5.41E+06	4.61E+06	1.82E+05	3.00E+03	0.	0.	0.	0.	0.	0.
EU-152	3.28E+04	1.80E+03	5.66E+02	3.76E-08	0.	0.	0.	0.	0.	0.	0.
EU-154	1.50E+07	1.72E+06	7.84E+05	1.68E-02	0.	0.	0.	0.	0.	0.	0.
EU-155	1.23E+05	6.00E+04	2.84E+07	0.	0.	0.	0.	0.	0.	0.	0.
GMFR	1.45E+01	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	1.57E+09	4.64E+08	2.00E+08	4.56E+05	1.71E+05	1.61E+05	1.47E+05	1.43E+05	1.12E+05	4.18E+04	2.18E+04

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.

TABLE A.2.1b. Radioactivity Inventory--Once-Through Cycle--Growth Case 1, Curies(A)

Actinides

RADIONUCLIDES (B)	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
Cm-245	9.09E+02	9.09E+02	9.09E+02	8.73E+02	8.47E+02	5.99E+02	4.44E+02	1.47E+01	2.07E+01	0.	0.
Cm-244	7.99E+06	4.41E+05	2.05E+05	3.62E+02	1.75E+10	0.	0.	0.	0.	0.	0.
Cm-243	1.39E+04	4.71E+03	3.75E+03	4.62E+01	9.14E+06	0.	0.	0.	0.	0.	0.
Cm-242	4.00E+04	4.98E+04	3.63E+04	5.70E+03	5.43E+02	6.98E+06	0.	0.	0.	0.	0.
Am-243+Am-242	1.56E+05	1.96E+05	1.55E+05	1.50E+05	1.03E+05	9.97E+04	4.43E+04	1.49E+03	1.42E+01	0.	0.
Am-241	1.78E+07	2.54E+07	2.92E+07	1.34E+07	6.01E+06	1.06E+04	4.08E+02	1.48E+01	2.08E+01	0.	0.
Pu-242	9.98E+03	9.98E+03	9.98E+03	9.97E+03	9.96E+03	9.99E+03	9.99E+03	9.11E+03	4.31E+03	4.00E+03	1.60E+03
Pu-241	4.09E+08	2.97E+07	1.16E+07	8.74E+02	8.49E+02	6.00E+02	4.04E+02	1.48E+01	2.08E+01	0.	0.
Pu-240	3.68E+06	4.87E+06	3.66E+06	3.51E+06	3.44E+06	2.21E+06	1.43E+06	4.19E+04	1.30E+02	0.	0.
Pu-239	2.72E+06	2.72E+06	2.72E+06	2.69E+06	2.45E+06	2.37E+06	2.06E+06	6.46E+05	1.61E+05	1.87E+00	1.27E+06
Pu-238	1.09E+07	7.42E+06	6.45E+06	2.79E+05	6.40E+03	1.68E+05	0.	0.	0.	0.	0.
Pu-236	2.30E+01	1.21E+04	9.40E+07	0.	0.	0.	0.	0.	0.	0.	0.
Np-237+Pa-233	4.38E+03	5.13E+03	5.06E+03	1.02E+04	1.44E+04	1.59E+04	1.48E+04	1.46E+04	1.43E+04	1.35E+04	1.15E+04
U-238+Th-234+Pa-234	9.52E+03	9.52E+03	9.42E+03	9.52E+03	9.42E+03	9.52E+03	9.42E+03	9.52E+03	9.52E+03	9.52E+03	9.52E+03
U-236	1.76E+03	1.77E+03	1.77E+03	1.81E+03	1.86E+03	2.18E+03	2.43E+03	2.40E+03	2.40E+03	2.77E+03	2.73E+03
U-235+Th-231	4.29E+02	4.30E+02	4.30E+02	4.32E+02	4.45E+02	4.54E+02	4.76E+02	5.72E+02	6.07E+02	6.18E+02	6.18E+02
U-234	9.82E+03	1.11E+04	1.15E+04	1.36E+04	1.44E+04	1.36E+04	1.45E+04	1.24E+04	1.12E+04	5.78E+03	3.81E+03
U-233	7.54E+01	1.26E+00	1.49E+00	8.23E+00	2.13E+01	1.52E+02	4.16E+02	1.40E+03	2.70E+03	6.28E+03	6.08E+03
U-232	9.58E+01	4.94E+01	4.93E+01	9.90E+01	4.43E+03	0.	0.	0.	0.	0.	0.
Pa-231	2.39E+01	4.71E+01	5.64E+01	2.35E+00	4.46E+00	2.24E+01	4.47E+01	1.73E+02	2.45E+02	3.09E+02	3.09E+02
Th-230	1.67E+00	6.22E+00	8.17E+00	5.20E+01	1.12E+02	5.74E+02	1.13E+03	4.44E+03	7.10E+03	6.74E+03	4.12E+03
Th-229+7 DAUGHTERS	4.10E+03	4.20E+02	6.28E+02	1.39E+00	6.43E+00	2.31E+02	4.73E+02	9.44E+03	1.97E+04	9.04E+04	4.89E+04
Th-228+6 DAUGHTERS	4.77E+02	4.30E+02	3.45E+02	7.12E+00	5.44E+02	3.00E+03	7.49E+03	4.27E+02	9.28E+02	4.75E+01	9.48E+01
Ac-227+7 DAUGHTERS	3.82E+01	2.54E+00	3.42E+00	1.85E+01	3.73E+01	1.79E+02	4.40E+02	1.49E+03	2.04E+03	2.47E+03	2.47E+03
Th-232+7 DAUGHTERS	1.42E+06	1.25E+05	1.77E+05	1.23E+04	2.41E+04	1.46E+03	4.17E+03	1.02E+02	3.08E+02	2.04E+01	4.06E+01
Ra-226+5 DAUGHTERS	4.45E+02	4.50E+01	9.17E+01	2.89E+01	1.22E+02	2.03E+03	4.06E+03	4.76E+04	4.29E+04	4.07E+04	2.47E+04
Pb-210+2 DAUGHTERS	4.31E+03	1.30E+01	2.07E+01	1.41E+01	6.12E+01	1.02E+03	2.43E+03	1.48E+04	2.15E+04	2.04E+04	1.24E+04
TOTAL	3.47E+04	6.97E+07	5.01E+07	2.01E+07	1.22E+07	4.75E+06	4.41E+06	7.02E+05	3.45E+05	1.64E+05	1.29E+05

A. VALUES LESS THAN 1.0E+10 HAVE BEEN DESIGNATED AS ZERO.
 B. Th-229, 7 DAUGHTERS ARE RA-225, AC-225, FR-221, AT-217, RI-213, PB-209 AND TL-200 IS 9% OF Th-229 AND PO-213 IS 9% OF Th-229.
 Th-228, 6 DAUGHTERS ARE RA-224, RN-220, PO-216, PB-212, RI-212 AND TL-208 IS 3% OF Th-228 AND PO-213 IS 6% OF Th-228.
 AC-227, 7 DAUGHTERS ARE Th-227, RA-223, RN-219, PO-215, PB-211, RI-211 AND TL-207.
 Th-232, 2 DAUGHTERS ARE RA-228 AND AC-228.
 Pa-226, 3 DAUGHTERS ARE RN-222, PO-214, PB-214, BI-214 AND Pa-214.
 Pb-210, 2 DAUGHTERS ARE BI-210 AND PO-210.

NOTE. IN ACCOUNTING FOR THE ACTIVITY IN THIS MANNER, BRANCHING DECAY IN THE CASE OF TL-208 (3%) = PO-212 (6%), AND TL-209 (4%) = PO-211 (4%) WERE COUNTED AS A SINGLE DAUGHTER IN EACH CASE. MINOR BRANCHING (1% OR LESS) WAS IGNORED.

TABLE A.2.2a. Radioactivity Inventory--Once-Through Cycle--Growth Case 2, Curies(A)

Fission and Activation Products

MAJOR RADIONUCLIDES	YEAR			BENLDTG TIME (YEARS BEYOND 1975)							
	2000	2050	2070	300	500	5000	10000	50000	100000	500000	1000000
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H-3	7.20E+06	1.06E+04	3.43E+05	4.03E+05	0.	0.	0.	0.	0.	0.	0.
C-14	1.61E+04	3.43E+04	3.82E+04	3.64E+04	3.03E+04	2.11E+04	1.15E+04	4.17E+01	2.17E+01	0.	0.
Mn-54	7.49E+02	5.57E+07	0.	0.	0.	0.	0.	0.	0.	0.	0.
Fe-55	7.06E+06	3.40E+03	1.45E+01	0.	0.	0.	0.	0.	0.	0.	0.
Co-60	1.31E+07	4.68E+04	3.36E+04	0.	0.	0.	0.	0.	0.	0.	0.
Ni-63	4.06E+04	1.03E+05	1.03E+05	1.02E+05	1.02E+05	1.37E+05	1.41E+05	9.26E+04	6.01E+04	1.88E+03	2.46E+01
Ni-63	4.37E+06	1.41E+07	1.30E+07	6.10E+05	1.01E+04	1.18E+09	0.	0.	0.	0.	0.
Se-79	4.60E+03	1.65E+04	1.64E+04	1.64E+04	1.43E+04	1.36E+04	1.08E+04	4.46E+03	5.47E+03	7.99E+01	3.48E+01
Kr-85	4.62E+07	1.73E+07	4.41E+06	2.36E+05	0.	0.	0.	0.	0.	0.	0.
Rb-87	7.35E+01	4.12E+01	8.12E+01	4.12E+01	4.12E+01	4.12E+01	4.12E+01	4.12E+01	4.12E+01	4.12E+01	4.12E+01
Sr-90+Y-90	1.73E+04	1.49E+04	1.03E+09	4.66E+04	2.07E+01	0.	0.	0.	0.	0.	0.
Zr-93	7.23E+04	7.41E+04	7.41E+04	7.41E+04	7.41E+04	7.79E+04	7.78E+04	7.43E+04	7.46E+04	6.20E+04	4.92E+04
Nb-93M	1.95E+04	7.31E+04	7.64E+04	7.42E+04	7.42E+04	7.40E+04	7.78E+04	7.44E+04	7.07E+04	6.21E+04	4.93E+04
Tc-99	2.51E+05	6.05E+05	6.05E+05	6.04E+05	6.03E+05	9.95E+05	4.45E+05	4.13E+05	4.35E+05	1.14E+05	2.23E+04
Ru-106+Rh-106	7.04E+06	1.25E+01	1.29E+07	0.	0.	0.	0.	0.	0.	0.	0.
Pd-107	1.93E+03	4.40E+03	4.40E+03	4.40E+03	4.40E+03	4.40E+03	4.40E+03	4.40E+03	4.40E+03	4.40E+03	4.40E+03
Ag-110M	3.88E+02	9.08E+09	0.	0.	0.	0.	0.	0.	0.	0.	0.
Cd-113M	7.46E+04	3.01E+04	1.12E+04	2.11E+05	0.	0.	0.	0.	0.	0.	0.
Sr-125+Y-125M	4.85E+06	6.40E+03	3.40E+01	0.	0.	0.	0.	0.	0.	0.	0.
Sr-126+Zr-126	2.79E+04	4.48E+04	6.48E+04	6.66E+04	6.44E+04	6.46E+04	4.24E+04	4.73E+04	3.24E+04	2.00E+03	6.57E+01
I-129	4.04E+02	1.55E+03	1.55E+03	1.55E+03	1.45E+03	1.55E+03	1.45E+03	1.45E+03	1.44E+03	1.52E+03	1.49E+03
Cs-134	7.32E+07	5.42E+03	6.93E+00	0.	0.	0.	0.	0.	0.	0.	0.
Cs-135	4.16E+03	1.26E+04	1.26E+04	1.26E+04	1.26E+04	1.26E+04	1.26E+04	1.26E+04	1.26E+04	1.26E+04	1.00E+04
Cs-137+Ba-137	2.42E+09	2.48E+09	1.57E+09	1.33E+05	1.29E+00	0.	0.	0.	0.	0.	0.
Ce-144+Pr-144	7.32E+05	3.38E+04	0.	0.	0.	0.	0.	0.	0.	0.	0.
Pm-147	4.13E+07	5.48E+04	2.47E+02	0.	0.	0.	0.	0.	0.	0.	0.
Sr-151	1.98E+07	7.52E+07	3.00E+07	1.18E+06	2.21E+04	3.26E+10	0.	0.	0.	0.	0.
Eu-152	9.41E+04	3.15E+04	9.43E+03	6.59E+07	0.	0.	0.	0.	0.	0.	0.
Eu-154	5.69E+07	2.74E+07	1.16E+07	2.69E+01	1.07E+10	0.	0.	0.	0.	0.	0.
Eu-155	4.79E+05	3.45E+01	1.43E+02	0.	0.	0.	0.	0.	0.	0.	0.
OTHER	2.73E+01	2.22E+10	0.	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	4.40E+09	6.28E+09	2.66E+09	3.01E+06	1.07E+06	1.01E+06	9.79E+05	4.44E+05	7.02E+05	2.62E+05	1.37E+05

A. VALUES LESS THAN 1.0E+10 HAVE BEEN DESIGNATED AS ZERO.

TABLE A.2.2b. Radioactivity Inventory--Once-Through Cycle--Growth Case 2, Curies(A)

Actinides

RADIOISOTOPES (A)	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	300	1000	5000	10000	50000	100000	300000	1000000
CM-244	7.31E+03	4.03E+03	8.02E+03	7.75E+03	7.63E+03	5.31E+03	7.09E+03	1.22E+02	1.88E+00	0.	0.
CM-244	1.34E+07	7.90E+06	3.67E+06	6.48E-01	3.13E-09	0.	0.	0.	0.	0.	0.
CM-243	7.37E+04	5.75E+04	3.73E+04	5.65E+00	1.12E+04	0.	0.	0.	0.	0.	0.
CM-242	1.50E+05	3.22E+05	2.94E+05	4.61E+04	4.71E+03	5.64E+05	0.	0.	0.	0.	0.
AM-243+AM-243	7.27E+05	1.27E+06	1.26E+06	1.22E+06	1.16E+06	8.11E+05	9.15E+05	1.77E+04	1.48E+02	0.	0.
AM-242+AM-242	7.88E+05	7.84E+05	7.15E+05	1.12E+05	1.15E+04	1.37E+04	0.	0.	0.	0.	0.
AM-241	4.22E+07	1.58E+08	1.62E+08	8.79E+07	3.65E+07	7.09E+04	7.42E+03	1.52E+02	1.44E+00	0.	0.
PU-242	3.09E+04	7.41E+04	7.41E+04	7.41E+04	7.40E+04	7.35E+04	7.38E+04	6.77E+04	6.17E+04	2.97E+04	1.19E+04
PU-241	1.01E+09	4.44E+08	1.74E+08	7.76E+03	7.45E+03	5.32E+03	7.40E+03	1.22E+02	1.44E+00	0.	0.
PU-240	8.97E+06	2.12E+07	2.12E+07	2.03E+07	1.93E+07	1.24E+07	7.47E+06	1.27E+05	7.53E+02	0.	0.
PU-239	7.80E+06	1.38E+07	1.34E+07	1.36E+07	1.44E+07	1.20E+07	1.85E+07	3.80E+06	4.22E+05	9.54E+00	6.46E+06
PU-238	3.61E+07	6.51E+07	5.57E+07	2.44E+06	5.47E+04	1.36E+04	0.	0.	0.	0.	0.
PU-236	1.80E+02	3.10E+01	2.39E+03	0.	0.	0.	0.	0.	0.	0.	0.
NP-237+PA-233	1.22E+04	3.31E+04	3.52E+04	6.95E+04	8.48E+04	1.03E+04	1.03E+05	1.02E+05	1.00E+05	8.79E+04	7.48E+04
U-238+Th-234+Pa-234m	1.90E+04	4.54E+04	4.54E+04	4.54E+04	4.54E+04	4.54E+04	4.54E+04	4.54E+04	4.54E+04	4.54E+04	4.54E+04
U-236	4.34E+03	1.08E+04	1.06E+04	1.09E+04	1.12E+04	1.30E+04	1.45E+04	1.46E+04	1.46E+04	1.64E+04	1.62E+04
U-235+Th-231	4.57E+07	1.48E+08	1.49E+08	1.70E+03	1.71E+03	1.81E+03	1.92E+03	2.01E+03	2.59E+03	2.64E+03	2.64E+03
U-234	1.96E+04	5.83E+04	6.17E+04	7.98E+04	8.17E+04	8.10E+04	8.01E+04	7.72E+04	4.56E+04	3.16E+04	1.92E+04
U-233	1.75E+00	4.90E+00	4.76E+00	5.07E+01	1.77E+02	9.84E+02	2.66E+03	9.78E+03	1.76E+04	4.10E+04	3.97E+04
U-232	5.15E+02	5.45E+02	4.89E+02	9.02E+00	7.42E+02	0.	0.	0.	0.	0.	0.
PA-231	4.11E+01	2.20E+00	2.56E+00	9.39E+00	1.86E+01	8.87E+01	1.74E+02	7.78E+02	1.08E+03	1.32E+03	1.32E+03
TH-230	3.21E+00	2.40E+01	3.40E+01	2.83E+02	6.06E+02	3.38E+03	6.47E+03	2.71E+04	4.18E+04	3.79E+04	2.11E+04
TH-229+7 DAUGHTERS	4.19E+03	1.91E+01	3.66E+01	8.15E+00	4.70E+01	1.49E+03	7.48E+03	6.58E+04	1.29E+05	3.29E+05	3.19E+05
TH-228+6 DAUGHTERS	2.14E+03	3.92E+03	3.93E+03	6.48E+01	5.70E+01	2.02E+02	4.40E+02	2.45E+01	5.40E+01	2.81E+00	5.61E+00
AC-227+7 DAUGHTERS	4.38E+01	1.16E+01	1.52E+01	7.36E+01	1.49E+02	7.10E+02	1.40E+03	9.77E+03	8.42E+03	1.06E+04	1.06E+04
TH-232+3 DAUGHTERS	3.47E+06	8.83E+05	8.94E+05	7.05E+04	1.45E+03	8.67E+03	1.89E+02	1.14E+01	2.36E+01	1.21E+00	2.40E+00
RA-226+4 DAUGHTERS	4.69E+02	1.92E+00	3.52E+00	1.47E+02	6.44E+02	1.19E+04	7.11E+04	1.43E+05	2.53E+05	2.28E+05	1.27E+05
FR-210+3 DAUGHTERS	4.87E+03	4.13E+01	8.81E+01	7.09E+01	3.42E+02	5.96E+03	1.58E+04	4.15E+04	1.27E+05	1.14E+05	4.33E+04
TOTAL	1.11E+09	7.13E+08	4.33E+08	1.26E+08	7.78E+07	2.61E+07	1.01E+07	4.50E+06	1.49E+06	9.75E+05	7.52E+05

 A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.
 B. TH-229, 7 DAUGHTERS ARE RA-225, AC-225, FR-221, AT-217, RI-213, PB-209 AND TL-209 VS 9% OF TH-229 AND PD-213 IS 91% OF TH-229.
 TH-229, 6 DAUGHTERS ARE RA-224, RN-220, PD-216, PB-212, RI-212 AND TL-208 IS 34% OF TH-229 AND PD-212 VS 64% OF TH-229.
 AC-227, 7 DAUGHTERS ARE TH-227, RA-223, RN-219, PD-215, PB-211, RI-211 AND TL-207.
 TH-232, 3 DAUGHTERS ARE RA-228 AND AC-228.
 RA-226, 4 DAUGHTERS ARE RN-222, PD-218, PB-214, BI-214 AND PN-214.
 PB-210, 2 DAUGHTERS ARE BI-210 AND PD-210.

NOTE. IN ACCOUNTING FOR THE ACTIVITY IN THIS MANNER, BRANCHING DECAY IN THE CASE OF TL-208 (34%) = PD-212 (64%), AND TL-209 (9%) = PD-211 (91%) WERE COUNTED AS A SINGLE DAUGHTER IN EACH CASE. MINOR BRANCHING (1% OR LESS) WAS IGNORED.

TABLE A.2.3a. Radioactivity Inventory--Once-Through Cycle--Growth Case 3, Curies(A)

Fission and Activation Products

MAJOR RADIOISOTOPES	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2080	2090	2070	900	1000	9000	10000	90000	100000	900000	1000000
M-3	5.24E+06	1.22E+07	3.95E+06	4.64E+04	0.	0.	0.	0.	0.	0.	0.
C-14	2.59E+04	1.99E+05	1.98E+05	1.89E+05	1.74E+05	1.10E+05	4.09E+04	2.76E+02	1.13E+00	0.	0.
MN-54	4.78E+02	1.15E+02	6.29E+06	0.	0.	0.	0.	0.	0.	0.	0.
FE-55	4.84E+06	2.93E+06	1.23E+04	0.	0.	0.	0.	0.	0.	0.	0.
CO-60	2.29E+07	2.11E+07	1.41E+06	0.	0.	0.	0.	0.	0.	0.	0.
NY-99	9.27E+04	7.22E+05	7.22E+05	7.20E+05	7.17E+05	6.92E+05	4.43E+05	4.19E+05	3.44E+05	9.52E+03	1.25E+02
NY-93	1.32E+07	4.23E+07	7.35E+07	3.41E+06	7.00E+04	6.58E+04	0.	0.	0.	0.	0.
SE-79	1.03E+04	4.26E+04	8.35E+04	8.22E+04	4.78E+04	7.83E+04	7.09E+04	4.25E+04	2.25E+04	4.01E+02	1.95E+00
KR-85	1.09E+08	2.27E+08	6.30E+07	3.59E+04	0.	0.	0.	0.	0.	0.	0.
RB-87	4.09E+01	4.08E+00	4.09E+00	4.08E+00	4.08E+00	4.08E+00	4.08E+00	4.08E+00	4.08E+00	4.08E+00	4.08E+00
SR-90+Y-90	2.72E+09	1.19E+10	7.24E+09	3.28E+05	1.06E+00	0.	0.	0.	0.	0.	0.
ZR-93	4.90E+04	3.92E+05	3.92E+05	3.92E+05	3.92E+05	3.91E+05	2.00E+05	2.13E+05	3.74E+05	3.11E+05	2.07E+05
NR-93M	2.81E+04	2.39E+05	3.73E+05	3.92E+05	3.92E+05	3.92E+05	2.91E+05	2.83E+05	3.75E+05	3.11E+05	2.07E+05
TC-99	2.11E+09	2.04E+08	3.03E+06	3.03E+06	2.03E+06	2.99E+06	2.99E+06	2.97E+06	2.18E+06	5.83E+05	1.12E+05
RU-104+PM-106	4.93E+06	1.03E+08	1.05E+00	0.	0.	0.	0.	0.	0.	0.	0.
PD-107	2.93E+04	2.31E+04	2.31E+04	2.31E+04	2.31E+04	2.31E+04	2.30E+04	2.30E+04	2.28E+04	2.20E+04	2.09E+04
AG-110M	4.90E+02	4.93E+01	1.03E+07	0.	0.	0.	0.	0.	0.	0.	0.
CO-113M	1.18E+05	2.11E+04	1.14E+05	2.18E+04	0.	0.	0.	0.	0.	0.	0.
SR-124+TE-125M	9.01E+06	2.46E+06	2.39E+04	0.	0.	0.	0.	0.	0.	0.	0.
SN-126+PR-128	4.24E+04	2.35E+05	3.39E+05	3.30E+05	3.23E+05	3.24E+05	2.73E+05	2.77E+05	1.66E+05	1.05E+04	3.30E+02
I-129	9.79E+02	7.78E+03	7.74E+03	7.78E+03	7.74E+03	7.78E+03	7.78E+03	7.77E+03	7.75E+03	7.63E+03	7.47E+03
CS-134	4.91E+07	1.91E+07	2.22E+04	0.	0.	0.	0.	0.	0.	0.	0.
CS-135	7.64E+03	6.31E+04	6.31E+04	6.31E+04	6.31E+04	6.30E+04	4.30E+04	4.24E+04	4.17E+04	3.62E+04	3.01E+04
CS-137+PA-137	2.79E+09	1.71E+10	1.08E+10	9.16E+05	4.27E+00	0.	0.	0.	0.	0.	0.
CE-140+PR-140	1.08E+06	2.40E+05	4.39E+03	0.	0.	0.	0.	0.	0.	0.	0.
PM-147	7.98E+07	3.99E+07	2.02E+05	0.	0.	0.	0.	0.	0.	0.	0.
SM-151	2.09E+07	1.96E+08	1.67E+08	6.60E+06	1.23E+05	1.42E+09	0.	0.	0.	0.	0.
EU-152	1.57E+05	3.70E+05	1.17E+05	7.75E+06	0.	0.	0.	0.	0.	0.	0.
EU-154	4.65E+07	2.57E+08	1.08E+08	2.52E+00	1.00E+09	0.	0.	0.	0.	0.	0.
EU-155	1.19E+06	3.52E+05	1.67E+02	0.	0.	0.	0.	0.	0.	0.	0.
OTMFR	2.22E+01	7.46E+00	6.32E+09	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	4.94E+09	2.99E+10	1.45E+10	1.65E+07	5.62E+06	5.07E+06	4.02E+06	4.19E+06	3.52E+06	1.31E+06	6.85E+05

A. VALUES LESS THAN 1.0E+10 HAVE BEEN DESIGNATED AS ZERO.

TABLE A.2.3b. Radioactivity Inventory--Once-Through Cycle--Growth Case 3, Curies(A)

Actinides

RADIOISOTOPES (A)	YEAR			GENEROIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
CM=243	4.84E+03	4.08E+04	4.08E+04	3.98E+04	3.78E+04	2.70E+04	1.78E+04	4.20E+02	9.36E+00	0.	0.
CM=244	2.03E+07	6.86E+07	3.19E+07	5.63E+00	2.72E+08	0.	0.	0.	0.	0.	0.
CM=243	7.95E+04	3.88E+05	2.42E+05	3.81E+01	7.43E+04	0.	0.	0.	0.	0.	0.
CM=242	2.11E+05	1.71E+06	1.46E+06	2.43E+05	2.41E+04	3.00E+04	0.	0.	0.	0.	0.
AM=243+AM=239	7.74E+05	6.39E+06	6.38E+06	6.15E+06	5.80E+06	4.09E+06	2.40E+06	4.07E+04	7.07E+02	0.	0.
AM=242+AM=242	4.72E+05	4.17E+06	3.81E+06	5.98E+05	6.11E+04	7.30E+04	0.	0.	0.	0.	0.
AM=241	6.14E+07	7.37E+08	8.84E+08	4.51E+08	2.83E+08	3.64E+05	1.79E+04	4.21E+02	9.38E+00	0.	0.
Pl=242	4.61E+04	3.72E+05	3.72E+05	3.72E+05	3.72E+05	3.69E+05	3.46E+05	3.40E+05	3.10E+05	1.49E+05	5.98E+04
PU=241	1.65E+09	4.39E+09	1.72E+09	3.45E+04	3.79E+04	2.71E+04	1.78E+04	6.51E+02	9.48E+00	0.	0.
PU=240	1.00E+07	1.06E+08	1.06E+08	1.02E+08	9.44E+07	6.02E+07	3.84E+07	6.76E+05	4.77E+03	0.	0.
Pl=239	9.28E+06	4.90E+07	6.90E+07	6.82E+07	6.73E+07	6.04E+07	4.28E+07	1.71E+07	4.12E+06	4.78E+01	3.24E+05
PU=238	4.37E+07	3.43E+08	3.11E+08	1.35E+07	3.39E+05	7.24E+04	0.	0.	0.	0.	0.
PU=236	3.16E+02	1.35E+02	1.04E+00	0.	0.	0.	0.	0.	0.	0.	0.
NP=237+PA=233	1.83E+04	1.40E+05	1.77E+05	3.22E+05	4.74E+05	5.18E+05	4.18E+05	4.11E+05	4.03E+05	4.02E+05	3.76E+05
U=238+TH=230+P=230+U=236	7.05E+04	2.27E+04	2.27E+05	2.27E+05	2.27E+05	2.27E+05	2.27E+05	2.27E+05	2.27E+05	2.27E+05	2.27E+05
U=236	6.60E+03	5.33E+04	5.34E+04	5.05E+04	5.40E+04	6.51E+04	7.25E+04	8.11E+04	8.72E+04	9.22E+04	8.10E+04
U=235+TH=231	1.07E+03	4.49E+03	8.05E+03	8.50E+03	8.47E+03	9.06E+03	9.41E+03	1.21E+04	1.30E+04	1.33E+04	1.32E+04
U=234	2.96E+04	2.40E+05	2.95E+05	3.94E+05	4.10E+05	4.07E+05	4.02E+05	3.47E+05	3.29E+05	1.58E+05	9.44E+04
U=233	2.60E+00	3.02E+01	3.75E+01	2.40E+02	4.71E+02	4.92E+03	1.07E+04	4.92E+04	8.86E+04	2.06E+05	1.99E+05
U=232	4.66E+02	3.12E+03	2.48E+03	5.17E+01	4.19E+01	0.	0.	0.	0.	0.	0.
PA=231	8.75E+01	9.43E+00	1.17E+01	4.52E+01	9.17E+01	4.43E+02	4.73E+02	3.40E+03	5.40E+03	6.63E+03	6.42E+03
TH=230	4.09E+00	9.44E+01	1.49E+02	1.34E+03	3.18E+03	1.69E+04	3.78E+04	1.28E+05	2.10E+05	1.90E+05	1.85E+05
TH=229+7 DAUGHTERS	7.35E+03	6.43E+01	1.19E+00	3.72E+01	2.88E+02	7.02E+03	2.88E+04	3.76E+05	4.07E+05	1.65E+06	1.40E+06
TH=228+6 DAUGHTERS	3.10E+03	2.23E+04	1.49E+04	3.72E+02	3.83E+00	1.01E+01	2.20E+01	1.23E+00	2.76E+00	1.41E+01	2.41E+01
AC=227+7 DAUGHTERS	7.48E+01	4.59E+01	6.43E+01	3.51E+02	7.40E+02	3.54E+03	4.09E+03	2.28E+04	4.32E+04	5.30E+04	5.30E+04
TH=232+2 DAUGHTERS	4.13E+06	2.12E+04	3.45E+04	3.38E+03	7.44E+03	4.33E+02	9.04E+02	9.49E+01	1.18E+00	6.05E+00	1.21E+01
RA=226+4 DAUGHTERS	1.13E+01	6.04E+00	1.24E+01	6.70E+02	3.42E+03	5.94E+04	1.48E+05	2.11E+05	1.27E+06	1.14E+06	6.43E+05
RA=210+2 DAUGHTERS	1.09E+02	1.17E+00	2.46E+00	3.17E+02	1.46E+03	2.97E+04	7.79E+04	4.09E+05	4.35E+05	5.71E+05	3.17E+05
TD7AL	1.81E+09	4.75E+09	3.66E+09	6.43E+08	3.75E+08	1.31E+08	4.46E+07	2.11E+07	4.09E+06	4.89E+06	3.77E+06

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.
 B. TH=229, 7 DAUGHTERS ARE RA=225, AC=225, FR=221, AT=217, OI=213, PB=209 AND TL=209 IS 9% OF TH=229 AND PO=213 IS 91% OF TH=229.
 TH=228, 6 DAUGHTERS ARE RA=220, RN=220, PU=216, PB=212, RI=212 AND TL=208 IS 74% OF TH=228 AND PO=212 IS 64% OF TH=228.
 AC=227, 7 DAUGHTERS ARE TH=227, RA=223, RN=219, PO=215, PR=211, PI=211 AND TL=207.
 TH=232, 2 DAUGHTERS ARE RA=228 AND AC=228.
 RA=226, 5 DAUGHTERS ARE RN=222, PO=214, PB=214, BI=214 AND PN=214.
 RB=210, 2 DAUGHTERS ARE RI=210 AND PO=210.
 NOTE. IN ACCOUNTING FOR THE ACTIVITY IN THIS MANNER, BRANCHING DECAY IN THE CASE OF TL=208 (36%) = PO=212 (64%), AND TL=209 (9%) = PO=213 (91%) WERE COUNTED AS A SINGLE DAUGHTER IN EACH CASE. MINOR BRANCHING (1% OR LESS) WAS IGNORED.

TABLE A.2.4a. Radioactivity Inventory--Once-Through Cycle--Growth Case 4, Curies(A)

Fission and Activation Products

MAJOR RADIONUCLIDES	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
H-3	3.98E+04	2.35E+07	7.61E+06	8.94E+04	0.	0.	0.	0.	0.	0.	0.
C-14	3.68E+02	2.63E+05	2.63E+05	2.50E+03	2.75E+03	1.45E+05	7.93E+04	6.70E+02	1.49E+00	0.	0.
Mn-56	3.57E+05	9.75E+02	5.17E+05	0.	0.	0.	0.	0.	0.	0.	0.
Fe-55	2.77E+03	1.23E+07	5.95E+04	0.	0.	0.	0.	0.	0.	0.	0.
Co-60	6.11E+04	6.44E+07	4.63E+06	0.	0.	0.	0.	0.	0.	0.	0.
Ni-49	1.41E+03	9.52E+05	9.52E+05	9.49E+03	9.69E+05	9.13E+05	4.74E+05	6.18E+05	4.01E+05	1.26E+04	1.65E+02
Ni-63	1.63E+05	1.16E+08	1.00E+08	4.70E+06	1.49E+05	9.07E+09	0.	0.	0.	0.	0.
Se-79	1.41E+02	1.08E+05	1.08E+05	1.08E+05	1.07E+05	1.03E+05	9.75E+04	4.77E+04	3.74E+04	5.26E+02	2.55E+00
Kr-85	7.52E+05	4.99E+04	1.27E+08	6.25E+04	0.	0.	0.	0.	0.	0.	0.
Rb-87	7.93E+03	5.75E+00	5.75E+00	5.35E+00	5.75E+00	5.75E+00	4.75E+00	5.75E+00	5.75E+00	5.35E+00	5.35E+00
Sr-90+Y-90	3.12E+07	1.43E+10	1.12E+10	5.02E+05	2.73E+00	0.	0.	0.	0.	0.	0.
Zr-93	7.65E+02	4.15E+05	5.15E+05	5.14E+05	5.14E+05	5.13E+05	4.12E+05	4.03E+05	4.01E+05	4.08E+05	3.24E+05
Nb-94M	4.40E+02	4.16E+05	4.77E+05	5.15E+05	5.15E+05	5.14E+05	4.13E+05	4.03E+05	4.02E+05	4.09E+05	3.25E+05
Tc-99	4.09E+03	3.99E+06	3.04E+06	3.98E+06	3.08E+06	3.92E+06	3.46E+06	3.78E+06	2.47E+06	7.67E+05	1.47E+05
Ru-104+Rh-104	1.40E+00	4.28E+06	8.41E+00	0.	0.	0.	0.	0.	0.	0.	0.
Pd-107	4.52E+01	3.04E+04	3.04E+04	3.04E+04	3.04E+04	3.04E+04	3.04E+04	3.02E+04	3.01E+04	2.89E+04	2.75E+04
Ag-110M	7.04E+07	7.64E+02	1.57E+06	0.	0.	0.	0.	0.	0.	0.	0.
Co-113M	7.40E+02	9.75E+05	2.14E+05	4.03E+04	0.	0.	0.	0.	0.	0.	0.
Sr-125+Y-125	3.69E+03	1.92E+07	1.14E+05	0.	0.	0.	0.	0.	0.	0.	0.
Sr-126+Zr-126	4.52E+02	4.41E+05	4.41E+05	4.40E+05	4.48E+05	4.48E+05	4.12E+05	3.12E+05	2.21E+05	1.38E+04	4.34E+02
I-129	1.53E+01	1.02E+04	1.02E+04	1.02E+04	1.42E+04	1.02E+04	1.02E+04	1.02E+04	1.02E+04	1.00E+04	9.42E+03
Cs-134	4.36E+03	1.17E+08	1.36E+05	0.	0.	0.	0.	0.	0.	0.	0.
Cs-135	1.04E+02	4.25E+04	8.25E+04	8.25E+04	4.25E+04	8.24E+04	4.23E+04	4.15E+04	4.06E+04	7.35E+04	6.55E+04
Cs-137+Ba-137	4.43E+07	2.40E+10	1.64E+10	1.39E+06	1.75E+01	0.	0.	0.	0.	0.	0.
Ce-144+Pr-144	1.43E+02	1.74E+06	3.19E+02	0.	0.	0.	0.	0.	0.	0.	0.
Pm-147	3.60E+04	1.73E+04	8.73E+05	0.	0.	0.	0.	0.	0.	0.	0.
Bm-151	4.49E+05	2.71E+08	2.31E+08	9.11E+06	1.70E+05	2.51E+09	0.	0.	0.	0.	0.
Eu-152	1.05E+03	7.10E+05	2.24E+05	1.49E+05	0.	0.	0.	0.	0.	0.	0.
Eu-154	4.46E+05	4.53E+04	1.91E+08	4.43E+00	1.76E+09	0.	0.	0.	0.	0.	0.
Eu-155	4.38E+01	2.30E+06	1.09E+03	0.	0.	0.	0.	0.	0.	0.	0.
Gd-157	4.20E+08	5.51E+01	4.70E+08	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	7.77E+07	4.40E+10	2.42E+10	2.26E+07	7.13E+06	6.66E+06	6.47E+06	5.50E+06	4.63E+06	1.72E+06	8.99E+05

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.

TABLE A.2.4b. Radioactivity Inventory--Once-Through Cycle--Growth Case 4, Curies(A)

Actinides

RADIOISOTOPES (a)	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
CM-245	4.79E+01	5.79E+04	5.50E+04	5.16E+04	4.85E+04	3.50E+04	2.33E+04	8.12E+02	1.23E+01	0.	0.
CM-244	1.27E+05	1.16E+08	5.40E+07	9.54E+00	4.40E+00	0.	0.	0.	0.	0.	0.
CM-243	6.63E+02	5.62E+05	3.77E+05	5.71E+01	1.13E+03	0.	0.	0.	0.	0.	0.
CM-242	1.42E+03	2.30E+06	2.10E+06	3.30E+03	3.47E+04	4.04E+04	0.	0.	0.	0.	0.
AM-243+MP-239	8.78E+03	8.38E+06	8.18E+06	8.07E+06	7.72E+06	5.37E+04	4.81E+06	4.10E+04	9.40E+02	0.	0.
AM-242+AM-242	4.02E+03	5.61E+06	5.12E+06	8.05E+05	8.32E+04	9.83E+04	0.	0.	0.	0.	0.
AM-241	1.34E+06	4.06E+08	1.04E+09	6.01E+08	2.70E+08	4.80E+05	2.45E+04	4.14E+02	1.23E+01	0.	0.
PU-242	5.96E+02	4.90E+04	4.90E+05	4.89E+05	4.49E+05	4.85E+05	4.81E+05	4.87E+05	4.08E+05	1.96E+05	7.47E+04
PU-241	1.42E+07	7.90E+09	3.13E+09	5.17E+04	4.66E+04	3.55E+04	2.43E+04	4.14E+02	1.23E+01	0.	0.
PU-240	2.93E+05	1.40E+08	1.40E+08	1.35E+08	1.34E+08	8.49E+07	4.08E+07	4.61E+05	4.99E+03	0.	0.
PU-239	1.92E+05	9.13E+07	9.13E+07	9.03E+07	8.81E+07	7.98E+07	4.06E+07	2.56E+07	5.45E+06	6.33E+01	4.29E+05
PU-238	5.65E+05	4.98E+08	4.98E+08	1.86E+07	4.03E+05	9.75E+04	0.	0.	0.	0.	0.
PU-236	1.15E+01	6.93E+02	5.35E+00	0.	0.	0.	0.	0.	0.	0.	0.
NP-237+PA-233	2.67E+02	2.07E+05	2.20E+05	4.16E+05	4.49E+05	6.82E+05	4.42E+05	4.74E+05	4.43E+05	5.82E+05	4.95E+05
U-238+Th-234+Pa-234m	4.73E+02	3.00E+05	3.00E+05	3.00E+05	3.00E+05	3.00E+05	3.00E+05	3.00E+05	3.00E+05	3.00E+05	3.00E+05
U-236	1.05E+02	6.98E+04	6.98E+04	7.13E+04	7.43E+04	8.50E+04	9.91E+04	1.19E+05	1.09E+05	1.04E+05	1.06E+05
U-235+Th-231	2.67E+01	1.10E+04	1.10E+04	1.11E+04	1.11E+04	1.18E+04	1.25E+04	1.44E+04	1.70E+04	1.73E+04	1.73E+04
U-234	4.03E+02	3.55E+04	3.41E+05	5.16E+05	5.44E+05	5.30E+05	4.34E+05	4.59E+05	4.29E+05	2.07E+05	1.26E+05
U-233	0.64E+02	3.91E+01	4.62E+01	3.04E+02	8.78E+02	6.45E+03	1.46E+04	6.14E+04	1.17E+05	2.71E+05	2.63E+05
U-232	4.00E+00	4.30E+03	3.37E+03	7.16E+01	5.41E+01	0.	0.	0.	0.	0.	0.
Pa-231	1.48E+02	1.22E+01	1.43E+01	5.75E+01	1.78E+02	5.75E+02	1.74E+03	4.70E+03	7.07E+03	8.67E+03	8.67E+03
Th-230	1.23E+01	1.12E+02	1.76E+02	1.69E+03	4.10E+03	2.20E+04	4.45E+04	1.57E+05	2.74E+05	2.44E+05	1.39E+05
Th-229+7 DAUGHTERS	4.50E+06	6.74E+01	1.30E+00	4.99E+01	2.48E+02	9.71E+03	4.74E+04	4.16E+05	4.42E+05	2.18E+06	2.11E+06
Th-228+6 DAUGHTERS	4.11E+01	3.06E+04	2.36E+04	5.15E+02	4.20E+00	1.32E+01	2.49E+01	1.59E+00	3.62E+00	1.86E+01	3.70E+01
Ac-227+7 DAUGHTERS	3.90E+04	4.96E+01	7.72E+01	4.44E+02	9.48E+02	4.60E+03	9.09E+03	3.76E+04	5.65E+04	6.90E+04	6.93E+04
Th-232+2 DAUGHTERS	2.95E+07	1.93E+04	3.46E+04	4.26E+03	9.49E+03	5.66E+02	1.24E+01	7.18E+01	1.53E+00	7.93E+00	1.58E+01
Ra-226+5 DAUGHTERS	4.69E+03	6.45E+00	1.34E+01	8.22E+02	4.24E+03	7.69E+04	2.43E+05	1.17E+06	1.66E+06	1.49E+06	8.32E+05
Po-210+2 DAUGHTERS	4.02E+04	1.21E+00	3.11E+00	3.44E+02	2.12E+03	3.85E+06	1.81E+05	5.43E+05	4.24E+05	7.46E+05	4.16E+05
TOTAL	1.67E+07	9.76E+09	4.90E+09	8.55E+08	4.47E+08	1.73E+04	1.86E+08	2.78E+07	1.12E+07	6.43E+06	4.97E+06

 A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.
 B. Th-229, 7 DAUGHTERS ARE RA-225, AC-229, FR-221, AT-217, RI-213, PB-209 AND TL-200 IS 9% OF TH-229 AND PO-213 IS 91% OF TH-229.
 Th-228, 6 DAUGHTERS ARE RA-224, RN-220, PO-216, PB-212, RI-212 AND TL-208 IS 3% OF TH-228 AND PN-212 IS 6% OF TH-228.
 AC-227, 7 DAUGHTERS ARE TH-227, RA-223, RN-219, PO-215, PR-211, RI-211 AND TL-207.
 Th-232, 2 DAUGHTERS ARE RA-228 AND AC-228.
 Ra-226, 5 DAUGHTERS ARE RN-222, PO-214, PB-210, BI-210 AND PN-210.
 Po-210, 2 DAUGHTERS ARE BI-210 AND PD-210.

NOTE. IN ACCOUNTING FOR THE ACTIVITY IN THIS MANNER, BRANCHING DECAY IN THE CASE OF Th-229 (3%) = PO-212 (64%), AND TL-209 (4%) = PO-211 (91%) WERE COUNTED AS A SINGLE DAUGHTER IN EACH CASE. MINOR BRANCHING (1% OR LESS) WAS IGNORED.

TABLE A.2.5a. Radioactivity Inventory--Once-Through Cycle--Growth Case 5, Curies(A)

Fission and Activation Products

MAJOR RADIOISOTOPES	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
H-3	4.98E+04	3.78E+07	1.23E+07	1.44E+03	0.	0.	0.	0.	0.	0.	0.
C-14	4.68E+02	3.59E+05	3.58E+05	3.41E+05	3.21E+05	1.98E+04	1.08E+05	4.49E+02	2.04E+00	0.	0.
MN-54	4.57E+05	1.75E+03	9.17E+05	0.	0.	0.	0.	0.	0.	0.	0.
FE-59	2.77E+03	2.28E+07	1.10E+05	0.	0.	0.	0.	0.	0.	0.	0.
CO-60	4.11E+04	1.34E+04	8.20E+06	0.	0.	0.	0.	0.	0.	0.	0.
NI-63	1.41E+03	1.30E+06	1.37E+06	1.29E+06	1.29E+06	1.24E+04	1.19E+06	4.01E+05	4.45E+05	1.71E+04	2.25E+02
NI-64	1.43E+05	1.43E+04	1.41E+06	6.98E+06	1.42E+05	1.27E+04	0.	0.	0.	0.	0.
BE-70	1.41E+02	1.47E+04	1.47E+05	1.47E+05	1.46E+05	1.40E+04	1.42E+05	4.45E+04	4.08E+04	7.15E+02	3.47E+00
KR-84	7.52E+05	7.50E+04	2.79E+08	1.02E+03	0.	0.	0.	0.	0.	0.	0.
RH-107	7.43E+03	7.26E+00	7.29E+00	7.26E+00	7.26E+00	7.26E+00	7.26E+00	7.26E+00	7.26E+00	7.26E+00	7.26E+00
SR-90+Y-90	4.12E+07	2.70E+10	1.45E+10	7.43E+05	3.40E+00	0.	0.	0.	0.	0.	0.
ZR-93	7.65E+02	6.99E+04	6.99E+05	6.99E+05	6.99E+05	6.97E+04	4.06E+05	4.43E+05	4.67E+05	5.55E+05	4.40E+05
NR-94	4.40E+02	4.42E+05	6.03E+05	6.99E+05	6.99E+05	6.98E+04	4.06E+05	4.44E+05	4.68E+05	5.55E+05	4.41E+05
TC-99	4.99E+03	4.42E+06	5.02E+06	5.41E+06	5.40E+06	5.33E+04	4.24E+06	4.49E+06	4.00E+06	1.04E+06	2.00E+05
HU-106+PM-106	1.80E+00	1.59E+07	1.43E+01	0.	0.	0.	0.	0.	0.	0.	0.
PD-107	4.52E+01	4.14E+04	4.14E+04	4.14E+04	4.14E+04	4.14E+04	4.13E+04	4.12E+04	4.10E+04	3.94E+04	3.75E+04
AG-110M	7.04E+07	1.48E+03	3.05E+06	0.	0.	0.	0.	0.	0.	0.	0.
CO-113M	7.90E+02	9.19E+05	3.42E+05	6.03E+04	0.	0.	0.	0.	0.	0.	0.
SR-125+TE-125M	4.69E+03	3.57E+07	2.11E+05	0.	0.	0.	0.	0.	0.	0.	0.
SM-126+P-126	4.52E+02	4.00E+04	6.90E+05	5.98E+05	5.98E+05	5.40E+04	4.40E+05	4.25E+05	3.00E+05	1.88E+04	5.90E+02
I-129	1.53E+01	1.39E+04	1.39E+04	1.39E+04	1.39E+04	1.39E+04	1.39E+04	1.39E+04	1.39E+04	1.36E+04	1.33E+04
CB-134	4.38E+03	2.21E+04	2.47E+05	0.	0.	0.	0.	0.	0.	0.	0.
CS-135	1.04E+02	1.12E+04	1.12E+05	1.12E+05	1.12E+05	1.12E+05	1.12E+05	1.11E+05	1.09E+05	9.97E+04	8.88E+04
CS-137+PA-137	4.43E+07	4.43E+10	2.41E+10	2.05E+06	1.04E+01	0.	0.	0.	0.	0.	0.
CF-140+PR-140	1.03E+02	3.36E+06	4.14E+02	0.	0.	0.	0.	0.	0.	0.	0.
PM-147	4.60E+04	3.16E+04	1.59E+06	0.	0.	0.	0.	0.	0.	0.	0.
SM-151	4.49E+05	3.78E+04	3.23E+08	1.27E+07	2.44E+05	3.50E+09	0.	0.	0.	0.	0.
EU-152	1.05E+03	1.15E+04	3.61E+05	2.40E+05	0.	0.	0.	0.	0.	0.	0.
EU-154	4.48E+05	7.11E+04	2.49E+08	6.96E+00	2.76E+09	0.	0.	0.	0.	0.	0.
EU-155	4.38E+01	4.35E+06	2.06E+03	0.	0.	0.	0.	0.	0.	0.	0.
OTMPR	4.20E+08	1.09E+02	9.31E+08	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	7.77E+07	6.41E+10	4.16E+10	3.14E+07	9.70E+06	9.05E+06	4.79E+06	7.48E+06	4.29E+06	2.34E+06	1.22E+06

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.

TABLE A.2.5b. Radioactivity Inventory--Once-Through Cycle--Growth Case 5, Curies(A)

Actinides

RADIOISOTOPES (a)	YEAR			GENUINE TYPE (YEAR BEYOND 1975)							
	2070	2080	2090	500	1000	3000	10000	30000	100000	300000	1000000
Cm-240	6.79E+01	7.29E+04	7.28E+04	7.04E+04	6.73E+04	6.42E+04	6.17E+04	5.91E+04	5.67E+04	5.44E+04	5.22E+04
Cm-244	1.27E+05	1.41E+04	8.40E+07	1.06E+01	7.14E+08	0.	0.	0.	0.	0.	0.
Cm-245	6.43E+02	6.57E+04	5.44E+05	8.42E+01	1.44E+03	0.	0.	0.	0.	0.	0.
Cm-247	1.42E+03	3.19E+04	2.91E+06	4.57E+05	4.67E+04	5.60E+04	0.	0.	0.	0.	0.
Am-241+Am-243	6.78E+03	1.14E+07	1.14E+07	1.10E+07	1.45E+07	7.32E+04	6.45E+06	1.54E+05	1.74E+03	0.	0.
Am-242+Am-244	6.42E+03	7.77E+04	7.09E+06	1.11E+06	1.10E+05	1.36E+03	0.	0.	0.	0.	0.
Am-241	1.34E+04	1.18E+06	1.20E+09	6.23E+08	3.90E+08	6.63E+05	3.20E+04	1.11E+03	1.67E+01	0.	0.
Pu-242	4.96E+02	6.67E+04	6.47E+05	6.47E+05	6.44E+05	6.61E+04	6.45E+04	6.20E+05	6.46E+05	2.67E+04	1.07E+05
Pu-241	1.42E+07	1.27E+10	4.94E+09	7.04E+04	6.76E+04	6.83E+04	6.18E+04	1.11E+03	1.47E+01	0.	0.
Pu-240	2.53E+04	1.91E+04	1.91E+08	1.83E+06	1.74E+06	1.16E+04	4.02E+07	1.74E+06	6.79E+03	0.	0.
Pu-239	1.92E+04	1.24E+04	1.24E+04	1.73E+04	1.21E+08	1.08E+04	4.05E+07	3.26E+07	7.01E+06	6.60E+01	5.42E+04
Pu-238	4.83E+04	6.07E+04	5.97E+08	2.60E+07	6.20E+05	1.45E+03	0.	0.	0.	0.	0.
Pu-236	1.15E+01	1.80E+04	1.90E+01	0.	0.	0.	0.	0.	0.	0.	0.
Np-237+Pa-233	2.67E+02	2.74E+04	2.95E+05	5.45E+05	7.73E+05	9.28E+05	9.29E+05	9.17E+05	9.13E+05	7.93E+05	6.74E+05
U-238+Th-234+Pa-234m	6.73E+02	4.06E+04	4.06E+05	4.06E+04	4.06E+05	4.06E+04	4.06E+05	4.06E+05	4.06E+05	4.06E+05	4.06E+05
U-238	1.05E+02	9.44E+04	9.47E+04	9.47E+04	9.44E+04	1.16E+05	1.29E+05	1.00E+05	1.08E+05	1.47E+05	1.45E+05
U-235+Th-231	2.47E+01	1.07E+04	1.07E+04	1.44E+04	1.49E+04	1.94E+04	1.88E+04	2.12E+04	2.29E+04	2.34E+04	2.38E+04
U-235	4.11E+02	4.71E+04	5.17E+05	6.95E+05	7.22E+05	7.16E+04	7.08E+04	6.07E+05	6.40E+05	2.80E+05	1.71E+05
U-233	4.64E+02	4.80E+01	6.03E+03	4.06E+02	1.18E+03	8.77E+03	1.45E+04	6.42E+04	1.59E+05	3.64E+05	3.56E+05
U-232	4.00E+00	6.04E+01	5.75E+03	1.01E+02	6.19E+01	0.	0.	0.	0.	0.	0.
Pa-231	1.48E+02	1.49E+01	1.91E+01	7.64E+01	1.44E+02	7.49E+02	1.42E+03	6.41E+03	9.41E+03	1.17E+04	1.17E+04
Th-230	1.23E+01	1.76E+02	2.21E+02	2.25E+03	4.42E+03	2.96E+04	6.47E+04	2.99E+05	3.70E+05	3.36E+05	1.86E+05
Th-229+ Daughters	4.54E+06	7.07E+01	1.62E+00	6.07E+01	3.40E+02	1.32E+04	6.09E+04	6.46E+05	1.16E+06	2.97E+06	2.86E+06
Th-228+ Daughters	4.11E+01	4.28E+04	3.62E+04	7.26E+02	5.02E+00	1.79E+01	3.42E+01	2.47E+00	4.02E+00	2.52E+01	5.02E+01
Ac-227+ Daughters	4.76E+00	6.05E+01	9.47E+01	5.89E+02	1.27E+03	6.15E+03	1.22E+04	6.05E+04	7.41E+04	9.34E+04	9.74E+04
Th-227+ Daughters	2.95E+07	2.27E+04	4.44E+04	5.70E+03	1.44E+02	7.46E+02	1.48E+01	1.01E+00	2.11E+00	1.04E+01	2.15E+01
Ra-226+ Daughters	6.69E+03	7.35E+00	1.45E+01	1.08E+03	5.47E+03	1.04E+04	2.73E+04	1.04E+06	2.24E+06	2.02E+06	1.13E+06
Pb-210+ Daughters	6.02E+04	1.43E+00	3.40E+00	5.02E+02	2.44E+03	5.19E+04	1.47E+05	7.50E+05	1.12E+06	1.01E+06	5.63E+05
TOTAL	1.67E+07	1.41E+10	7.48E+09	1.17E+09	6.79E+08	2.45E+04	1.72E+04	3.78E+07	1.42E+07	6.72E+06	6.75E+06

1. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.
 2. Th-229, 7 DAUGHTERS ARE Ra-225, Ac-224, Fr-221, At-217, Pt-213, Pb-209 AND Tl-209 TO 9% OF Th-229 AND Po-213 IS 91% OF Th-229.
 Th-228, 6 DAUGHTERS ARE Ra-224, Rn-220, Po-216, Pb-212, Bi-212 AND Tl-208 TO 74% OF Th-228 AND Pb-212 IS 64% OF Th-228.
 Ac-227, 7 DAUGHTERS ARE Th-227, Ra-223, Rn-219, Po-215, Pb-211, Bi-211 AND Tl-207.
 Th-227, 2 DAUGHTERS ARE Ra-223 AND Ac-223.
 Ra-226, 5 DAUGHTERS ARE Rn-222, Pb-218, Bi-214 AND Po-214.
 Pb-210, 2 DAUGHTERS ARE Bi-210 AND Po-210.

NOTE. IN ACCOUNTING FOR THE ACTIVITY IN THIS MANNER, BRANCHING DECAY IN THE CASE OF Th-209 (34%) = Po-212 (64%), AND Tl-209 (9%) = Pb-211 (91%) WERE COUNTED AS A SINGLE DAUGHTER IN EACH CASE. OTHER BRANCHING (1% OR LESS) WAS IGNORED.

TABLE A.2.6a. Radioactivity Inventory--Reprocessing Cycle--Growth Case 3 - 1990 Reprocessing Startup, Curies(A)

Fission and Activation Products

MAJOR RADIOISOTOPES	YEAR			GENERIC TIME (YEARS BEYOND 1975)							
	2000	2040	2170	500	1000	5000	10000	50000	100000	500000	1000000
	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----
H-3	7.44E+05	1.07E+06	6.37E+05	7.44E+05	0.	0.	0.	0.	0.	0.	0.
C-14	2.73E+04	1.90E+05	1.89E+05	1.80E+05	1.49E+05	1.04E+05	5.71E+04	4.53E+02	1.08E+00	0.	0.
MN-54	4.22E+02	1.27E+02	7.04E+06	0.	0.	0.	0.	0.	0.	0.	0.
FE-55	5.15E+06	2.10E+06	1.02E+04	0.	0.	0.	0.	0.	0.	0.	0.
CO-60	2.10E+07	1.94E+07	1.39E+06	0.	0.	0.	0.	0.	0.	0.	0.
NI-63	6.89E+04	4.28E+05	6.29E+05	6.29E+05	6.23E+05	4.01E+05	5.75E+05	4.07E+05	2.64E+05	4.28E+03	1.09E+02
NI-63	1.26E+07	7.28E+07	6.29E+07	2.95E+06	4.83E+04	5.69E+09	0.	0.	0.	0.	0.
SE-79	3.25E+03	7.88E+04	7.44E+04	7.41E+04	7.77E+04	7.04E+04	7.46E+04	4.21E+04	2.70E+04	3.81E+02	1.85E+00
KR-85	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
RB-87	3.60E+01	3.62E+00	3.62E+00	3.62E+00	3.62E+00	3.62E+00	3.62E+00	3.62E+00	3.62E+00	3.62E+00	3.62E+00
BR-90+Y-90	7.39E+04	1.00E+10	6.35E+09	2.86E+05	1.27E+00	0.	0.	0.	0.	0.	0.
ZR-93	3.82E+04	4.84E+05	3.84E+05	3.82E+05	3.84E+05	3.87E+05	3.87E+05	3.79E+05	3.71E+05	3.08E+05	2.45E+05
NB-93M	3.23E+04	3.37E+05	3.70E+05	3.88E+05	3.89E+05	3.95E+05	3.87E+05	3.80E+05	3.71E+05	3.09E+05	2.45E+05
TC-99	3.20E+05	3.01E+06	3.01E+06	3.00E+06	3.00E+06	2.96E+06	2.91E+06	2.85E+06	2.16E+06	5.78E+05	1.11E+05
RU-106+PM-106	3.64E+04	1.19E+06	1.22E+00	0.	0.	0.	0.	0.	0.	0.	0.
PO-107	2.24E+02	2.88E+04	2.88E+04	2.88E+04	2.88E+04	2.87E+04	2.87E+04	2.84E+04	2.85E+04	2.70E+04	2.61E+04
AG-110M	2.67E+01	1.00E+02	2.07E+07	0.	0.	0.	0.	0.	0.	0.	0.
CO-113M	4.64E+04	4.31E+05	2.35E+05	4.43E+04	0.	0.	0.	0.	0.	0.	0.
SB-125+TE-125M	5.46E+05	4.89E+06	2.77E+04	0.	0.	0.	0.	0.	0.	0.	0.
BN-126+RB-126	3.33E+04	3.91E+05	3.91E+05	3.90E+05	3.88E+05	3.78E+05	3.45E+05	2.77E+05	1.96E+05	1.23E+04	3.84E+02
I-129	2.32E+02	4.29E+03	8.29E+03	8.29E+03	8.29E+03	8.29E+03	8.29E+03	8.28E+03	8.26E+03	8.13E+03	7.96E+03
CB-134	3.74E+06	1.91E+07	2.22E+04	0.	0.	0.	0.	0.	0.	0.	0.
CS-135	2.37E+03	7.94E+04	7.94E+04	7.94E+04	7.94E+04	7.93E+04	7.92E+04	7.85E+04	7.76E+04	7.07E+04	6.30E+04
CS-137+BA-137	1.04E+09	1.71E+10	1.08E+10	9.13E+05	8.84E+00	0.	0.	0.	0.	0.	0.
CE-140+PR-144	8.62E+02	2.34E+05	4.24E+03	0.	0.	0.	0.	0.	0.	0.	0.
PM-147	4.24E+06	3.88E+07	1.96E+05	0.	0.	0.	0.	0.	0.	0.	0.
SM-151	9.45E+06	2.09E+04	1.75E+08	7.04E+06	1.31E+05	1.94E+09	0.	0.	0.	0.	0.
EU-152	3.06E+04	6.44E+05	2.03E+05	1.35E+05	0.	0.	0.	0.	0.	0.	0.
EU-154	2.05E+07	3.11E+04	1.31E+08	3.04E+00	1.21E+09	0.	0.	0.	0.	0.	0.
EU-155	2.53E+04	4.19E+05	1.98E+02	0.	0.	0.	0.	0.	0.	0.	0.
DTMFR	2.80E+00	6.72E+00	5.70E+09	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	1.86E+09	2.82E+10	1.79E+10	1.64E+07	5.25E+06	5.01E+06	4.87E+06	4.76E+06	3.51E+06	1.32E+06	6.98E+05

A. VALUES LESS THAN 1.0E+10 HAVE BEEN DESIGNATED AS ZERO.

TABLE A.2.6b. Radioactivity Inventory--Reprocessing Cycle--Growth Case 3 - 1990 Reprocessing Startup, Curies(A)

Actinides

RADIONUCLIDES (A)	YEAR			GENLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
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CN-245	1.03E+03	7.19E+05	7.18E+05	6.94E+05	6.65E+05	4.76E+05	3.13E+05	1.69E+04	1.65E+02	0.	0.
CN-244	4.81E+06	7.72E+04	3.99E+08	6.34E+01	3.66E+07	0.	0.	0.	0.	0.	0.
CN-243	2.11E+04	1.01E+06	6.50E+05	9.94E+01	1.96E+03	0.	0.	0.	0.	0.	0.
CN-242	7.02E+04	1.24E+07	1.13E+07	1.77E+06	1.81E+05	2.17E+03	0.	0.	0.	0.	0.
AM-243+NP-239	2.34E+05	2.99E+07	2.98E+07	2.47E+07	2.75E+07	1.91E+07	1.21E+07	3.24E+05	3.49E+03	0.	0.
AM-242+AM-242	1.71E+05	3.02E+07	2.75E+07	4.32E+06	4.02E+05	5.28E+03	0.	0.	0.	0.	0.
AM-241	1.25E+07	4.04E+04	4.15E+06	2.25E+08	1.02E+08	6.65E+05	3.14E+05	1.09E+04	1.65E+02	0.	0.
PU-242	3.09E+02	1.30E+05	1.30E+05	1.30E+05	1.41E+05	1.30E+05	1.28E+05	1.19E+05	1.09E+05	5.24E+04	2.10E+04
PU-241	1.15E+07	1.18E+09	4.43E+08	6.94E+05	6.66E+05	4.77E+05	3.13E+05	1.69E+04	1.65E+02	0.	0.
PU-240	9.72E+04	1.41E+07	1.71E+07	1.73E+07	1.45E+07	1.09E+07	6.45E+06	1.88E+05	4.03E+02	0.	0.
PU-239	6.17E+04	3.59E+04	3.57E+06	3.71E+06	3.46E+06	4.67E+06	4.07E+06	3.04E+06	4.08E+05	7.06E+00	4.78E+06
PU-238	3.21E+05	1.13E+04	9.41E+07	7.73E+06	4.04E+05	5.24E+03	0.	0.	0.	0.	0.
PU-236	6.26E+00	6.45E+04	4.09E+02	0.	0.	0.	0.	0.	0.	0.	0.
NP-237+PA-233	5.74E+03	2.72E+04	2.73E+05	3.54E+05	4.09E+05	4.53E+05	4.94E+05	4.90E+05	4.43E+05	3.89E+05	3.31E+05
U-238+Th-234+ Pa-234m	1.46E+02	1.47E+03	1.47E+03	1.47E+03	1.47E+03	1.47E+03	1.47E+03	1.47E+03	1.48E+03	1.49E+03	1.90E+03
U-236	2.46E+01	6.45E+02	6.55E+02	8.42E+02	1.10E+03	2.66E+03	3.01E+03	4.73E+03	5.76E+03	5.69E+03	5.61E+03
U-235+Th-231	5.10E+00	6.85E+01	6.44E+01	6.93E+01	7.41E+01	1.07E+02	1.45E+02	4.33E+02	5.03E+02	6.34E+02	6.34E+02
U-234	1.35E+02	1.09E+04	1.04E+04	9.16E+04	5.05E+04	5.83E+04	4.75E+04	4.15E+04	4.04E+04	1.50E+04	4.16E+03
U-233	9.78E+02	2.42E+01	3.24E+01	2.86E+02	7.14E+02	4.48E+03	4.19E+03	4.43E+04	7.40E+04	1.81E+05	1.76E+05
U-232	2.24E+00	4.73E+05	3.92E+05	7.27E+03	6.38E+01	0.	0.	0.	0.	0.	0.
PA-231	2.63E+01	7.92E+00	7.93E+00	4.13E+00	4.04E+00	1.14E+01	1.49E+01	1.07E+02	2.18E+02	3.17E+02	3.17E+02
TH-230	1.30E+00	1.46E+01	2.10E+01	1.50E+02	4.13E+02	2.34E+03	4.75E+03	1.02E+04	2.93E+04	2.07E+04	5.84E+03
TH-229+ DAUGHTERS	2.47E+04	2.49E+01	6.94E+01	4.37E+01	2.40E+02	5.92E+03	2.46E+04	2.78E+05	4.70E+05	1.46E+06	1.41E+06
TH-228+ DAUGHTERS	4.04E+01	3.37E+02	2.42E+06	5.46E+04	4.09E+02	2.28E+03	4.03E+03	5.71E+02	1.25E+01	6.68E+01	1.34E+00
AC-227+ DAUGHTERS	4.30E+01	4.14E+01	5.21E+01	6.48E+01	6.75E+01	9.11E+01	1.25E+02	4.20E+02	1.75E+03	2.54E+03	2.94E+03
TH-227+ DAUGHTERS	1.20E+06	3.04E+04	4.32E+06	4.37E+05	1.10E+04	9.77E+04	2.75E+03	2.45E+02	4.37E+02	2.86E+01	5.73E=01
RA-226+ DAUGHTERS	5.30E+02	1.94E+00	3.00E+00	7.20E+01	4.08E+02	8.27E+03	2.21E+04	1.16E+05	1.97E+05	1.25E+05	3.50E+04
FR-210+ DAUGHTERS	6.33E+03	4.87E+01	6.45E+01	3.36E+01	2.04E+02	4.14E+03	1.10E+04	4.79E+04	4.86E+04	6.23E+04	1.75E+04
TOTAL	3.59E+07	2.57E+09	1.43E+09	2.90E+08	1.43E+08	3.70E+07	2.44E+07	4.05E+06	2.16E+06	2.31E+06	2.01E+06

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.
 B. TH-229, 7 DAUGHTERS ARE PA-225, AC-225, FR-221, AT-217, RI-213, PB-209 AND TL-209 IS 9% OF TH-229 AND PO-213 IS 91% OF TH-229.
 TH-228, 6 DAUGHTERS ARE RA-224, RN-220, PO-216, PB-212, RI-212 AND TL-208 IS 34% OF TH-228 AND PO-212 IS 64% OF TH-228.
 AC-227, 7 DAUGHTERS ARE TH-227, PA-223, RN-219, PD-215, PR-211, RI-211 AND TL-207.
 TH-232, 2 DAUGHTERS ARE RA-225 AND AC-228.
 PA-226, 5 DAUGHTERS ARE RN-222, PO-214, PB-214, BI-214 AND PR-214.
 PB-210, 2 DAUGHTERS ARE RI-210 AND PO-210.

NOTE. IN ACCOUNTING FOR THE ACTIVITY IN THIS MANNER, BRANCHING DECAY IN THE CASE OF TL-208 (34%) = PO-212 (64%), AND TL-209 (9%) = PO-211 (91%) WERE COUNTED AS A SINGLE DAUGHTER IN EACH CASE. MINOR BRANCHING (1% OR LESS) WAS IGNORED.

TABLE A.2.7a. Radioactivity Inventory--Reprocessing Cycle--Growth Case 3 - 2010 Reprocessing Startup, Curies(A)

Fission and Activation Products

MAJOR RADIONUCLIDES	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	2040	2070	500	1000	5000	10000	50000	100000	300000	1000000
	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----
H-3	0.	1.90E+04	6.16E+05	7.25E+05	0.	0.	0.	0.	0.	0.	0.
C-14	0.	2.12E+09	2.11E+05	2.01E+05	1.09E+05	1.17E+09	6.98E+04	9.77E+02	1.20E+00	0.	0.
MN-54	0.	1.24E+07	6.92E+06	0.	0.	0.	0.	0.	0.	0.	0.
FE-55	0.	2.18E+04	1.06E+04	0.	0.	0.	0.	0.	0.	0.	0.
CO-60	0.	2.03E+07	1.06E+06	0.	0.	0.	0.	0.	0.	0.	0.
NI-63	0.	7.00E+05	7.00E+05	6.97E+05	6.00E+05	6.71E+05	6.22E+05	6.44E+05	2.95E+05	9.23E+03	1.22E+02
NI-64	0.	8.13E+07	7.00E+07	3.29E+06	7.43E+04	6.35E+09	0.	0.	0.	0.	0.
SE-79	0.	6.04E+04	6.74E+04	8.04E+04	8.00E+04	7.67E+04	7.27E+04	6.75E+04	2.79E+04	3.92E+02	1.90E+00
KR-85	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
Rb-87	0.	3.91E+00	3.91E+00	3.91E+00	3.91E+00	3.91E+00	3.91E+00	3.91E+00	3.91E+00	3.91E+00	3.91E+00
SR-90+Y-90	0.	1.13E+10	6.89E+09	3.10E+05	1.78E+00	0.	0.	0.	0.	0.	0.
ZR-93	0.	4.05E+05	4.05E+05	4.05E+05	4.05E+05	4.04E+05	4.03E+05	3.96E+05	3.87E+05	3.21E+05	2.55E+05
NB-93M	0.	3.51E+05	3.46E+05	4.05E+05	4.05E+05	4.04E+05	4.03E+05	3.96E+05	3.87E+05	3.22E+05	2.55E+05
TC-99	0.	3.01E+06	3.01E+06	3.00E+06	3.00E+06	2.96E+06	2.91E+06	2.85E+06	2.16E+06	9.78E+05	1.11E+05
RU-106+RH-106	0.	1.14E+06	1.22E+00	0.	0.	0.	0.	0.	0.	0.	0.
PO-107	0.	2.46E+04	2.06E+04	2.46E+04	2.46E+04	2.46E+04	2.46E+04	2.45E+04	2.43E+04	2.34E+04	2.23E+04
AG-110M	0.	9.66E+01	1.99E-07	0.	0.	0.	0.	0.	0.	0.	0.
CO-113M	0.	5.25E+05	1.95E+05	3.49E+04	0.	0.	0.	0.	0.	0.	0.
BB-125+TE-125M	0.	4.67E+04	2.76E+04	0.	0.	0.	0.	0.	0.	0.	0.
SN-124+SA-126	0.	3.49E+05	3.49E+05	3.48E+05	3.47E+05	3.38E+05	3.26E+05	2.87E+05	1.75E+05	1.10E+04	3.43E+02
I-129	0.	7.92E+03	7.92E+03	7.92E+03	7.92E+03	7.92E+03	7.92E+03	7.91E+03	7.89E+03	7.76E+03	7.61E+03
CB-134	0.	1.94E+07	2.26E+04	0.	0.	0.	0.	0.	0.	0.	0.
CS-135	0.	6.70E+04	6.70E+04	6.70E+04	6.70E+04	6.70E+04	6.69E+04	6.63E+04	6.55E+04	5.97E+04	5.32E+04
CB-137+PA-137	0.	1.70E+10	1.07E+10	9.09E+05	8.21E+00	0.	0.	0.	0.	0.	0.
CE-144+PR-144	0.	2.27E+05	4.16E+03	0.	0.	0.	0.	0.	0.	0.	0.
FM-147	0.	3.87E+07	1.96E+05	0.	0.	0.	0.	0.	0.	0.	0.
EM-151	0.	1.99E+04	1.49E+08	6.68E+06	1.25E+05	1.84E+09	0.	0.	0.	0.	0.
EU-152	0.	4.88E+05	1.54E+05	1.02E+05	0.	0.	0.	0.	0.	0.	0.
EU-154	0.	2.82E+04	1.19E+06	2.76E+00	1.10E+09	0.	0.	0.	0.	0.	0.
EU-155	0.	4.21E+05	2.07E+02	0.	0.	0.	0.	0.	0.	0.	0.
OTHER	0.	6.42E+00	5.65E+09	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	0.	2.89E+10	1.40E+10	1.64E+07	5.02E+06	9.07E+06	6.92E+06	6.79E+06	3.53E+06	1.33E+06	7.05E+05

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.

TABLE A.2.7b. Radioactivity Inventory--Reprocessing Cycle--Growth Case 3 - 2010 Reprocessing Startup, Curies(A)

Actinides

RADIOISOTOPES (h)	YEAR			GEOLGIC TIME (YEARS SINCE 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
CM=245	0.	1.89E+04	1.89E+05	1.83E+05	1.75E+05	1.25E+04	8.20E+04	2.28E+03	4.34E+01	0.	0.
CM=244	0.	2.76E+04	1.29E+06	2.27E+01	1.09E+07	0.	0.	0.	0.	0.	0.
CM=243	0.	9.80E+05	3.76E+05	5.70E+01	1.13E+03	0.	0.	0.	0.	0.	0.
CM=242	0.	4.06E+04	3.70E+06	5.81E+05	5.04E+04	7.12E+04	0.	0.	0.	0.	0.
AM=243+MP=239	0.	1.12E+07	1.12E+07	1.08E+07	1.03E+07	7.16E+04	4.95E+04	1.21E+05	1.31E+03	0.	0.
AM=242+AM=242	0.	9.88E+06	9.07E+06	1.42E+06	1.05E+05	1.73E+03	0.	0.	0.	0.	0.
AM=241	0.	6.53E+04	6.74E+08	3.31E+08	1.09E+08	3.77E+04	8.27E+04	2.28E+03	4.35E+01	0.	0.
PU=242	0.	5.15E+07	5.18E+03	5.42E+03	5.49E+03	5.45E+03	5.00E+03	5.02E+03	4.98E+03	2.21E+03	8.84E+02
PU=241	0.	6.83E+07	2.41E+07	1.83E+05	1.76E+05	1.26E+04	8.25E+04	2.28E+03	4.35E+01	0.	0.
PU=240	0.	1.77E+06	2.16E+06	2.80E+06	2.28E+06	1.51E+06	9.07E+05	1.40E+04	8.90E+01	0.	0.
PU=239	0.	7.41E+05	7.44E+05	7.92E+05	8.49E+05	1.22E+04	1.45E+06	7.70E+05	1.03E+05	2.24E+00	1.52E+06
PU=238	0.	6.37E+06	6.01E+06	1.17E+06	1.29E+05	1.72E+03	0.	0.	0.	0.	0.
PU=236	0.	1.62E+00	2.79E-02	0.	0.	0.	0.	0.	0.	0.	0.
NP=237+PA=233	0.	1.89E+05	1.97E+05	3.07E+05	3.00E+05	4.93E+05	4.43E+05	4.08E+05	4.01E+05	3.87E+05	3.29E+05
U=238+TH=234+ PA=234M	0.	1.84E+03	1.44E+03	1.84E+03	1.84E+03	1.44E+03	1.84E+03	1.84E+03	1.84E+03	1.84E+03	1.84E+03
U=236	0.	4.79E+02	4.80E+02	5.05E+02	5.02E+02	7.57E+02	9.11E+02	1.15E+03	1.19E+03	1.17E+03	1.15E+03
U=235+TH=231	0.	6.70E+01	6.70E+01	6.76E+01	6.84E+01	7.66E+01	8.99E+01	1.24E+02	2.25E+02	2.34E+02	2.38E+02
U=234	0.	2.55E+03	2.90E+03	6.01E+03	7.19E+03	7.26E+03	7.17E+03	6.27E+03	5.70E+03	2.27E+03	1.02E+03
U=233	0.	7.75E+00	1.60E+01	2.15E+02	6.15E+02	4.33E+03	9.08E+03	4.21E+04	7.76E+04	1.80E+05	1.75E+05
U=232	0.	3.56E+01	2.95E+01	5.92E+01	4.80E+03	0.	0.	0.	0.	0.	0.
PA=231	0.	8.22E+00	8.23E+00	8.42E+00	8.71E+00	1.09E+01	1.01E+01	4.27E+01	8.65E+01	1.19E+02	1.19E+02
TH=230	0.	9.26E+01	5.31E+01	6.80E+01	9.05E+01	3.02E+02	6.76E+02	2.04E+03	3.71E+03	2.92E+03	1.21E+03
TH=229+7 DAUGHTERS	0.	6.54E+02	2.45E+01	2.98E+01	1.88E+02	6.56E+03	2.91E+04	2.77E+05	5.47E+05	1.45E+06	1.01E+06
TH=228+6 DAUGHTERS	0.	4.59E+02	2.12E+02	4.26E+00	3.07E+02	9.35E+04	2.09E+03	1.42E+02	2.76E+02	1.42E+01	2.84E+01
AC=227+7 DAUGHTERS	0.	3.09E+01	4.74E+01	6.70E+01	6.07E+01	8.75E+01	1.13E+02	1.29E+02	6.92E+02	9.54E+02	9.53E+02
TH=232+2 DAUGHTERS	0.	1.61E+05	4.06E+06	2.84E+05	6.77E+05	4.01E+04	8.07E+04	5.66E+03	1.18E+02	6.10E+02	1.22E+01
RA=226+5 DAUGHTERS	0.	4.70E+00	7.39E+00	5.82E+01	1.49E+02	1.27E+03	3.02E+03	1.27E+04	2.24E+04	1.74E+04	7.28E+03
PB=210+2 DAUGHTERS	0.	1.01E+00	2.02E+00	2.77E+01	7.47E+01	6.37E+02	1.41E+03	7.24E+03	1.12E+04	8.78E+03	3.64E+03
TOTAL	0.	1.03E+09	8.22E+08	3.49E+08	1.83E+08	1.10E+07	7.46E+06	1.72E+06	1.33E+06	2.06E+06	1.93E+06

A. VALUES LESS THAN 1.0E+10 HAVE BEEN DESIGNATED AS ZERO.
B. TH=229, 7 DAUGHTERS ARE PA=225, AC=225, FN=221, AT=217, RT=213, PB=209 AND TL=206 VS 9% OF TH=229 AND PD=213 IS 91% OF TH=229.
TH=228, 6 DAUGHTERS ARE RA=224, RN=220, PD=216, PB=212, RI=212 AND TL=204 IS 74% OF TH=228 AND PD=219 VS 60% OF TH=228.
AC=227, 7 DAUGHTERS ARE TH=227, RA=223, RN=219, PD=215, PB=211, BI=211 AND TL=207.
TH=232, 2 DAUGHTERS ARE RA=228 AND AC=228.
PA=226, 5 DAUGHTERS ARE RN=222, PD=214, PB=214, BI=214 AND PN=214.
PB=210, 2 DAUGHTERS ARE BI=210 AND PD=210.

NOTE. IN ACCOUNTING FOR THE ACTIVITY IN THIS MANNER, BRANCHING DECAY IN THE CASE OF TL=208 (34%) - PD=212 (64%), AND TL=209 (9%) - PD=211 (91%) WERE COUNTED AS A SINGLE DAUGHTER IN EACH CASE. MINOR BRANCHING (1% OR LESS) WAS IGNORED.

TABLE A.2.8a. Radioactivity Inventory--Reprocessing Cycle--Growth Case 4 - 2000 Reprocessing Startup, Curies(A)

Fission and Activation Products

MAJOR RADIONUCLIDES	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----
H=3	4.00E+03	3.75E+06	1.22E+06	1.43E+04	0.	0.	0.	0.	0.	0.	0.
C=10	4.75E+02	2.61E+05	2.60E+05	2.08E+05	2.43E+05	1.44E+05	7.46E+04	6.34E+02	1.08E+00	0.	0.
MN=54	3.57E+05	1.06E+03	5.90E+05	0.	0.	0.	0.	0.	0.	0.	0.
FE=55	2.77E+03	1.01E+07	4.89E+04	0.	0.	0.	0.	0.	0.	0.	0.
CO=60	4.11E+04	6.05E+07	4.34E+06	0.	0.	0.	0.	0.	0.	0.	0.
NI=59	1.01E+03	4.44E+05	8.64E+05	8.61E+05	8.47E+05	8.28E+05	7.43E+05	5.41E+05	3.64E+05	1.14E+04	1.50E+02
NI=63	1.83E+04	1.05E+04	9.00E+07	4.23E+06	9.41E+04	4.17E+09	0.	0.	0.	0.	0.
SE=79	4.56E+02	1.04E+05	1.04E+05	1.04E+05	1.03E+05	9.48E+04	9.47E+04	6.12E+04	3.59E+04	5.06E+02	2.45E+00
KR=85	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
RB=87	4.22E+06	4.90E+00	4.90E+00	4.90E+00	4.90E+00	4.90E+00	4.90E+00	4.90E+00	4.90E+00	4.90E+00	4.90E+00
BP=90+Y=90	1.66E+04	1.43E+10	9.98E+09	4.09E+05	1.09E+00	0.	0.	0.	0.	0.	0.
ZR=93	4.32E+01	5.18E+04	5.18E+05	5.18E+05	5.18E+05	5.17E+05	4.16E+05	5.06E+05	4.95E+05	4.11E+05	3.26E+05
NB=93M	3.54E+01	4.22E+04	4.24E+05	5.18E+05	5.18E+05	5.17E+05	5.16E+05	5.07E+05	4.95E+05	4.12E+05	3.27E+05
TC=99	4.20E+00	3.95E+04	3.95E+06	3.95E+06	3.84E+06	3.89E+06	3.42E+06	3.45E+06	2.44E+06	7.60E+05	1.66E+05
RU=106+PM=106	3.11E+03	1.06E+07	1.04E+01	0.	0.	0.	0.	0.	0.	0.	0.
PO=107	2.01E+02	3.58E+04	3.54E+04	3.48E+04	3.48E+04	3.58E+04	3.47E+04	3.46E+04	3.44E+04	3.40E+04	3.24E+04
AG=110M	3.75E+10	1.30E+03	2.44E+06	0.	0.	0.	0.	0.	0.	0.	0.
CD=113M	2.76E+02	1.12E+06	4.17E+05	7.88E+04	0.	0.	0.	0.	0.	0.	0.
SB=125+TE=125M	2.42E+01	2.36E+07	1.00E+05	0.	0.	0.	0.	0.	0.	0.	0.
SN=126+RB=126	3.47E+01	4.93E+04	4.93E+05	4.92E+05	4.00E+05	4.77E+05	4.41E+05	3.69E+05	2.07E+05	1.55E+04	4.85E+02
I=129	1.51E+01	1.07E+04	1.07E+04	1.07E+04	1.07E+04	1.07E+04	1.07E+04	1.07E+04	1.07E+04	1.05E+04	1.03E+04
CS=134	2.85E+00	1.20E+04	1.39E+05	0.	0.	0.	0.	0.	0.	0.	0.
CS=135	5.56E+02	9.77E+04	9.77E+04	9.77E+04	9.76E+04	9.76E+04	9.74E+04	9.45E+04	9.44E+04	8.70E+04	7.75E+04
CS=137+PA=137	2.36E+04	2.59E+10	1.63E+10	1.38E+06	1.44E+01	0.	0.	0.	0.	0.	0.
CE=144+PR=144	7.59E+06	1.81E+06	3.37E+02	0.	0.	0.	0.	0.	0.	0.	0.
PM=147	1.02E+01	1.48E+08	8.49E+05	0.	0.	0.	0.	0.	0.	0.	0.
SM=151	2.60E+02	2.83E+04	2.41E+08	9.53E+06	1.78E+05	2.62E+09	0.	0.	0.	0.	0.
EU=152	4.61E+01	1.15E+06	3.62E+05	2.41E+05	0.	0.	0.	0.	0.	0.	0.
EU=154	3.04E+02	5.36E+04	2.25E+08	5.24E+00	2.08E+09	0.	0.	0.	0.	0.	0.
EU=155	2.87E+02	2.85E+06	1.35E+03	0.	0.	0.	0.	0.	0.	0.	0.
OTMFR	2.39E+04	4.81E+01	4.14E+08	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	2.06E+05	4.35E+10	2.69E+10	2.24E+07	7.08E+06	6.61E+06	6.03E+06	5.68E+06	4.62E+06	1.74E+06	9.20E+05

VALUES LESS THAN 1.0E+10 HAVE BEEN DESIGNATED AS ZERO.

TABLE A.2.8b. Radioactivity Inventory--Reprocessing Cycle--Growth Case 4 - 2000 Reprocessing Startup, Curies(A)

Actinides

RADIONUCLIDES (A)	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
CM=245	2.55E+02	6.03E+05	6.02E+05	5.42E+05	5.48E+05	3.99E+05	2.42E+05	9.16E+03	1.78E+02	0.	0.
CM=244	6.76E+01	9.23E+04	4.29E+04	7.58E+01	3.46E+07	0.	0.	0.	0.	0.	0.
CM=243	3.43E+01	1.25E+06	8.12E+05	1.23E+02	2.03E+03	0.	0.	0.	0.	0.	0.
CM=242	7.58E+01	1.10E+07	1.01E+07	1.58E+06	1.62E+05	1.94E+03	0.	0.	0.	0.	0.
AM=243+MP=239	4.68E+00	2.63E+07	2.63E+07	2.53E+07	2.42E+07	1.68E+07	1.07E+07	2.25E+05	3.07E+03	0.	0.
AM=242+AM=242	3.42E+00	2.69E+07	2.46E+07	3.86E+06	3.04E+05	4.71E+03	0.	0.	0.	0.	0.
AM=241	7.29E+02	4.09E+04	5.94E+08	3.11E+08	1.00E+08	6.49E+05	2.43E+05	9.16E+03	1.79E+02	0.	0.
PU=242	3.07E+00	1.13E+04	1.14E+04	1.20E+04	1.22E+04	1.21E+04	1.20E+04	1.12E+04	1.02E+04	4.91E+03	1.97E+03
PU=241	9.68E+04	1.76E+04	6.94E+07	5.83E+05	5.49E+05	4.00E+05	2.63E+05	9.16E+03	1.79E+02	0.	0.
PU=240	1.22E+03	4.77E+04	6.04E+06	6.93E+06	6.49E+06	4.37E+04	2.62E+04	4.73E+04	2.57E+02	0.	0.
PU=239	9.02E+02	1.20E+04	1.21E+06	1.33E+06	1.40E+06	2.01E+04	2.09E+04	1.28E+04	4.21E+05	4.90E+00	3.32E+06
PU=238	3.14E+03	2.06E+07	1.91E+07	3.33E+06	3.01E+05	4.67E+03	0.	0.	0.	0.	0.
PU=236	9.80E+02	2.73E+01	2.10E+01	0.	0.	0.	0.	0.	0.	0.	0.
NP=237+PA=233	1.42E+01	3.41E+05	3.49E+05	4.53E+05	5.71E+05	5.90E+05	4.01E+05	4.18E+05	5.74E+05	9.04E+05	4.31E+05
U=238+TH=234+PA=234M	2.11E+00	2.46E+03	2.46E+03	2.46E+03	2.06E+03	2.06E+03	2.06E+03	2.06E+03	2.06E+03	2.47E+03	2.47E+03
U=236	3.18E+01	8.13E+02	8.17E+02	8.49E+02	9.04E+02	1.62E+03	2.12E+03	2.45E+03	2.55E+03	2.52E+03	2.78E+03
U=235+TH=231	7.70E+02	8.78E+01	8.78E+01	8.87E+01	9.02E+01	1.06E+02	1.33E+02	1.25E+02	4.25E+02	4.54E+02	4.54E+02
U=234	1.52E+00	4.38E+03	5.09E+03	1.49E+04	1.42E+04	1.40E+04	1.01E+04	1.23E+04	1.03E+04	5.19E+03	1.90E+03
U=233	1.40E+04	1.66E+01	3.14E+01	3.49E+02	9.07E+02	5.78E+03	1.20E+04	4.24E+04	1.01E+05	2.36E+05	2.26E+05
U=232	1.52E+02	8.20E+01	6.86E+01	1.38E+00	1.12E+02	0.	0.	0.	0.	0.	0.
PA=231	7.96E+06	1.09E+01	1.09E+01	1.11E+01	1.15E+01	1.05E+01	1.01E+01	1.23E+01	1.49E+02	2.27E+02	2.27E+02
TH=230	6.68E+05	3.81E+01	3.90E+01	7.46E+01	1.45E+02	7.72E+02	1.42E+03	4.09E+03	9.28E+03	6.93E+03	2.41E+03
TH=229+ DAUGHTERS	1.06E+04	1.69E+01	5.33E+01	5.07E+01	2.00E+02	2.92E+03	3.43E+04	3.42E+05	7.41E+05	1.89E+06	1.84E+06
TH=228+ DAUGHTERS	1.94E+02	6.70E+02	4.93E+02	9.90E+00	8.06E+02	1.60E+03	3.41E+03	2.70E+02	4.22E+02	2.49E+01	4.96E+01
AC=227+ DAUGHTERS	2.10E+07	4.65E+01	6.57E+01	6.65E+01	9.21E+01	1.16E+02	1.43E+02	6.49E+02	1.27E+03	1.82E+03	1.81E+03
TH=232+ DAUGHTERS	1.58E+10	5.79E+06	4.58E+06	4.90E+05	1.16E+04	6.87E+04	1.45E+03	9.25E+03	2.07E+02	1.07E+01	2.12E+01
RA=226+ DAUGHTERS	2.50E+06	3.68E+00	5.65E+00	5.29E+01	1.44E+02	2.73E+03	7.09E+03	1.48E+04	4.42E+04	4.16E+04	1.44E+04
PB=210+ DAUGHTERS	3.21E+07	4.34E+01	1.58E+00	2.52E+01	9.18E+01	1.36E+03	3.84E+03	1.43E+04	2.01E+04	2.04E+04	7.22E+03
TOTAL	1.03E+05	1.80E+09	1.18E+09	3.55E+04	1.75E+08	2.57E+07	1.74E+07	1.14E+06	1.97E+06	2.72E+06	2.83E+06

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.
B. TH=229, 7 DAUGHTERS ARE RA=225, AC=225, FR=221, AT=217, RT=213, PB=209 AND TL=208 IS 9% OF TH=229 AND PD=213 IS 91% OF TH=229.
TH=228, 6 DAUGHTERS ARE RA=224, RN=220, PU=216, PB=212, RT=212 AND TL=208 IS 34% OF TH=228 AND PD=212 IS 64% OF TH=228.
AC=227, 7 DAUGHTERS ARE TH=227, RA=223, RN=219, PU=215, PA=211, PI=211 AND TL=207.
TH=232, 2 DAUGHTERS ARE RA=228 AND AC=228.
PA=226, 5 DAUGHTERS ARE RN=222, PN=219, PB=214, BI=214 AND PD=214.
PB=210, 2 DAUGHTERS ARE RT=210 AND PD=210.

NOTE. IN ACCOUNTING FOR THE ACTIVITY IN THIS MANNER, BRANCHING DECAY IN THE CASE OF TL=208 (34%) = PD=212 (64%), AND TL=209 (9%) = PD=211 (91%) WERE COUNTED AS A SINGLE DAUGHTER IN EACH CASE. MINOR BRANCHING (1% OR LESS) WAS IGNORED.

TABLE A.2.9a. Radioactivity Inventory--Reprocessing Cycle--Growth Case 5 - 2000 Reprocessing Startup, Curies(A)

Fission and Activation Products

MAJOR RADIOISOTOPES	YEAR			GEOLOGIC TIME (YEARS BEYOND 1995)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----
H=3	4.00E+03	6.06E+06	1.06E+06	2.31E+04	0.	0.	0.	0.	0.	0.	0.
C=14	4.75E+02	3.93E+04	3.42E+05	3.35E+05	3.15E+05	1.94E+05	1.06E+05	4.03E+02	2.00E+00	0.	0.
MN=54	4.57E+05	1.88E+03	1.05E+04	0.	0.	0.	0.	0.	0.	0.	0.
FE=55	2.77E+03	1.79E+07	8.68E+04	0.	0.	0.	0.	0.	0.	0.	0.
CO=60	4.11E+04	1.05E+08	7.97E+06	0.	0.	0.	0.	0.	0.	0.	0.
NI=69	1.41E+03	1.17E+04	1.16E+06	1.16E+06	1.16E+06	1.12E+06	1.07E+06	7.46E+05	4.90E+05	1.54E+04	2.02E+02
NI=63	1.83E+05	1.45E+04	1.25E+08	5.87E+06	1.76E+05	1.13E+04	0.	0.	0.	0.	0.
SE=79	4.56E+02	1.41E+04	1.41E+05	1.40E+05	1.00E+05	1.34E+04	1.27E+05	4.38E+04	4.86E+04	6.85E+02	3.32E+00
KR=85	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
RB=87	4.22E+06	6.59E+00	6.59E+00	6.59E+00	6.59E+00	6.59E+00	4.49E+00	4.49E+00	4.49E+00	6.59E+00	6.59E+00
SR=90+Y=90	1.66E+04	2.40E+10	1.06E+10	6.59E+05	2.03E+00	0.	0.	0.	0.	0.	0.
ZR=93	4.32E+01	7.00E+04	7.00E+05	7.00E+05	7.00E+05	6.99E+05	6.07E+05	4.04E+05	6.49E+05	5.56E+05	4.01E+05
NR=93M	3.54E+01	5.07E+04	6.45E+05	7.00E+05	7.01E+05	6.99E+05	6.08E+05	4.05E+05	6.49E+05	5.56E+05	4.02E+05
TC=99	4.40E+00	5.37E+04	5.37E+06	5.36E+06	5.45E+06	5.28E+06	5.19E+06	4.45E+06	3.86E+06	1.07E+06	1.98E+05
RU=106+PM=106	3.11E+03	2.03E+07	2.09E+01	0.	0.	0.	0.	0.	0.	0.	0.
PD=107	2.41E+02	4.94E+04	4.04E+04	4.94E+04	4.04E+04	4.94E+04	4.04E+04	4.07E+04	4.90E+04	4.71E+04	4.08E+04
AG=110M	4.75E+10	2.45E+04	5.05E+06	0.	0.	0.	0.	0.	0.	0.	0.
CO=113M	2.78E+02	1.78E+06	6.61E+05	1.25E+03	0.	0.	0.	0.	0.	0.	0.
BR=125+Y=125M	2.42E+01	4.43E+07	2.42E+05	0.	0.	0.	0.	0.	0.	0.	0.
BN=126+SR=126	3.47E+01	6.79E+04	6.79E+05	6.77E+05	6.74E+05	6.56E+04	6.74E+05	4.40E+05	3.40E+05	2.13E+04	6.67E+02
I=129	1.51E+01	1.46E+04	1.46E+04	1.46E+04	1.46E+04	1.46E+04	1.46E+04	1.46E+04	1.46E+04	1.43E+04	1.40E+04
CS=134	2.45E+00	2.20E+04	2.55E+05	0.	0.	0.	0.	0.	0.	0.	0.
CS=135	4.56E+02	1.34E+05	1.34E+05	1.34E+05	1.34E+05	1.34E+05	1.34E+05	1.33E+05	1.31E+05	1.20E+05	1.07E+05
CS=137+BA=137	2.36E+04	3.81E+10	2.40E+10	2.04E+06	1.07E+01	0.	0.	0.	0.	0.	0.
CE=144+PR=144	7.59E+06	3.74E+06	6.11E+02	0.	0.	0.	0.	0.	0.	0.	0.
PM=147	1.92E+01	3.08E+08	1.46E+06	0.	0.	0.	0.	0.	0.	0.	0.
BN=151	2.60E+02	3.95E+08	3.39E+08	1.34E+07	2.40E+05	3.69E+09	0.	0.	0.	0.	0.
EU=152	4.61E+01	1.89E+06	5.06E+05	3.95E+05	0.	0.	0.	0.	0.	0.	0.
EU=154	3.44E+02	4.36E+08	3.52E+08	8.18E+00	3.95E+09	0.	0.	0.	0.	0.	0.
EU=155	2.87E+02	5.34E+06	2.53E+03	0.	0.	0.	0.	0.	0.	0.	0.
OTHER	2.39E+09	8.71E+01	7.49E+08	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	2.96E+05	6.42E+10	3.05E+10	3.12E+07	9.42E+06	8.98E+06	4.72E+06	7.44E+06	6.27E+06	2.36E+06	1.25E+06

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.

TABLE A.2.9b. Radioactivity Inventory--Reprocessing Cycle--Growth Case 5 - 2000 Reprocessing Startup, Curies(A)

Actinides

RADIONUCLIDES (B)	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----
Cm-245	2.55E+02	9.87E+09	9.55E+05	9.23E+05	6.85E+05	6.33E+05	4.16E+05	1.25E+04	2.19E+02	0.	0.
Cm-246	6.76E+01	1.59E+09	7.49E+08	1.30E+02	6.29E+07	0.	0.	0.	0.	0.	0.
Cm-243	7.43E-01	1.96E+06	1.27E+06	1.92E+02	3.40E+03	0.	0.	0.	0.	0.	0.
Cm-242	7.58E-01	1.76E+07	1.61E+07	2.52E+06	2.88E+05	3.09E+03	0.	0.	0.	0.	0.
Am-243+MP-239	4.68E+00	4.08E+07	4.07E+07	3.93E+07	3.75E+07	2.61E+07	1.46E+07	4.02E+05	4.77E+03	0.	0.
Am-242+Am-242	7.42E+00	4.29E+07	3.91E+07	6.14E+06	6.28E+05	7.51E+03	0.	0.	0.	0.	0.
Am-241	7.29E+02	6.37E+08	8.14E+08	4.30E+08	1.04E+08	9.82E+05	4.17E+05	1.26E+04	2.20E+02	0.	0.
Pu-242	7.07E+00	2.29E+04	2.31E+04	2.41E+04	2.04E+04	2.43E+04	2.40E+04	2.53E+04	2.04E+04	9.81E+03	3.93E+03
Pu-241	9.68E+04	3.92E+04	1.44E+04	9.24E+05	6.87E+05	6.34E+04	4.17E+05	1.08E+04	2.20E+02	0.	0.
Pu-240	1.22E+03	7.35E+06	9.41E+06	1.11E+07	1.06E+07	7.01E+06	4.20E+05	6.94E+04	4.12E+02	0.	0.
Pu-239	9.02E+02	1.80E+06	1.91E+06	2.00E+06	2.25E+06	3.68E+04	4.89E+06	2.29E+06	6.49E+05	7.55E+00	5.11E+06
Pu-238	7.14E+03	3.22E+07	3.00E+07	5.27E+06	6.07E+05	7.44E+03	0.	0.	0.	0.	0.
Pu-236	9.80E-02	6.11E+04	6.26E+02	0.	0.	0.	0.	0.	0.	0.	0.
Np-237+Pa-233	1.42E-01	4.22E+05	4.33E+05	5.76E+05	6.44E+05	7.66E+05	7.27E+05	7.21E+05	7.09E+05	6.58E+05	5.99E+05
U-238+Th-234+ Pa-234m	2.11E+00	3.35E+03	3.35E+03	3.35E+03	3.35E+03	3.35E+03	3.35E+03	3.35E+03	3.19E+03	3.36E+03	3.36E+03
U-236	7.18E-01	1.01E+03	1.02E+03	1.13E+03	1.30E+03	2.30E+03	3.10E+03	4.27E+03	4.28E+03	4.23E+03	4.17E+03
U-235+Th-231	7.70E-02	1.15E+02	1.15E+02	1.17E+02	1.19E+02	1.42E+02	1.44E+02	4.09E+02	6.34E+02	6.79E+02	6.78E+02
U-234	1.52E+00	5.55E+03	7.30E+03	2.20E+04	2.73E+04	2.76E+04	2.73E+04	2.45E+04	2.14E+04	7.73E+03	2.74E+03
U-233	1.40E-04	1.92E+01	3.74E+01	4.35E+02	1.15E+03	7.06E+03	1.25E+04	7.27E+04	1.32E+05	3.06E+05	2.97E+05
U-232	1.52E-02	2.16E+04	2.06E+04	4.13E+02	7.36E+00	0.	0.	0.	0.	0.	0.
Pa-231	7.96E+06	1.36E+01	1.36E+01	1.39E+01	1.04E+01	1.45E+01	2.49E+01	1.19E+02	2.35E+02	3.39E+02	3.39E+02
Th-230	6.68E+05	4.66E+01	4.77E+01	9.93E+01	2.19E+02	1.15E+03	2.27E+03	4.19E+03	1.00E+04	1.00E+04	3.91E+03
Th-229+ DAUGHTERS	1.06E-08	1.84E+01	6.13E+01	6.26E+01	7.46E+02	1.15E+04	4.90E+04	4.70E+05	9.62E+05	2.46E+06	2.39E+06
Th-228+ DAUGHTERS	1.94E-02	1.41E+05	1.49E+05	2.97E+03	2.01E+01	2.08E+03	4.81E+03	3.21E+02	6.78E+02	3.52E-01	7.02E-01
Ac-227+ DAUGHTERS	2.10E-07	5.51E+01	8.04E+01	1.11E+02	1.15E+02	1.08E+02	1.09E+02	4.22E+02	1.88E+03	2.71E+03	2.71E+03
Th-232+ DAUGHTERS	1.58E+10	7.94E+06	5.58E+06	6.08E-05	1.05E-04	4.93E-04	2.06E-03	1.33E-02	2.91E-02	1.51E-01	3.01E-01
Ra-226+ DAUGHTERS	2.50E+06	4.21E+00	6.61E+00	6.71E+01	2.41E+02	4.03E+03	1.06E+04	9.91E+04	4.05E+04	6.22E+04	2.11E+04
Pb-210+ DAUGHTERS	3.21E-07	9.20E-01	1.81E+00	3.17E+01	1.25E+02	2.02E+03	5.30E+03	2.79E+04	4.22E+04	3.11E+04	1.05E+04
TOTAL	1.03E+05	2.96E+09	1.45E+09	4.99E+08	2.44E+08	3.09E+07	2.73E+07	4.20E+06	2.69E+06	3.56E+06	3.30E+06

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.
 B. Th-229, 7 DAUGHTERS ARE RA-225, AC-224, FM-221, AT-217, PT-213, PB-209 AND TL-200 VS 9% OF Th-229 AND PD-213 IS 91% OF Th-229.
 Th-228, 6 DAUGHTERS ARE PA-224, RN-223, PD-216, PB-212, BI-212 AND TL-208 IS 34% OF Th-228 AND PD-212 IS 66% OF Th-228.
 Ac-227, 7 DAUGHTERS ARE TH-227, RA-223, RN-219, PD-215, PH-211, PI-211 AND TL-207.
 Th-232, 2 DAUGHTERS ARE RA-228 AND AC-228.
 Pa-226, 5 DAUGHTERS ARE RN-222, PU-214, PB-214, BI-214 AND PN-214.
 Pb-210, 2 DAUGHTERS ARE BI-210 AND PD-210.

NOTE. IN ACCOUNTING FOR THE ACTIVITY IN THIS MANNER, BRANCHING DECAY IN THE CASE OF TL-204 (34%) = PD-212 (64%), AND TL-209 (9%) = PD-231 (91%) WERE COUNTED AS A SINGLE DAUGHTER IN EACH CASE. MINOR BRANCHING (1% OR LESS) WAS IGNORED.

A.3 HEAT GENERATION RATE TABLES

The tables of heat generation rates (A.3.1a through A.3.9b) appear in the same format as those for radioactivity inventory.

TABLE A.3.1a. Heat Generation Rates--Once-Through Cycle--Growth Case 1, Watts(A)

Fission and Activation Products

MAJOR RADIONUCLIDES	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
H-3	7.61E+01	2.16E+00	7.00E+01	0.	0.	0.	0.	0.	0.	0.	0.
C-10	1.83E+00	1.42E+00	1.41E+00	1.73E+00	1.62E+00	1.00E+00	5.27E-01	2.43E-03	1.03E-05	0.	0.
MN-54	1.23E+00	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
FE-55	1.20E+03	1.96E+03	9.92E+06	0.	0.	0.	0.	0.	0.	0.	0.
CO-60	6.04E+04	8.32E+01	5.99E+00	0.	0.	0.	0.	0.	0.	0.	0.
NI-69	1.10E+00	1.10E+00	1.10E+00	1.10E+00	1.09E+00	1.05E+00	1.01E+00	7.73E-01	4.63E-01	1.45E-02	1.91E-04
NI-63	4.08E+02	3.49E+02	3.00E+02	1.41E+01	3.27E-01	0.	0.	0.	0.	0.	0.
SE-79	9.96E+01	9.96E+01	9.95E+01	9.91E+01	9.86E+01	9.85E+01	9.86E+01	5.25E+01	3.03E+01	4.83E+03	2.35E+05
KR-85	3.53E+04	1.03E+03	3.96E+02	1.92E+09	0.	0.	0.	0.	0.	0.	0.
RR-87	8.52E+05	8.52E+05	8.92E+05	8.92E+05	8.92E+05	8.52E+05	8.92E+05	8.92E+05	8.92E+05	8.52E+05	8.52E+05
BR-90+Y-90	2.26E+06	6.99E+05	4.02E+05	1.81E+01	8.93E-05	0.	0.	0.	0.	0.	0.
ZR-93	1.48E+00	1.48E+00	1.48E+00	1.48E+00	1.48E+00	1.48E+00	1.48E+00	1.48E+00	1.48E+00	1.18E+00	9.34E-01
NB-93M	2.89E+00	4.24E+00	4.94E+00	4.38E+00	4.96E+00	4.37E+00	4.96E+00	4.37E+00	4.18E+00	3.48E+00	2.76E+00
TC-99	1.67E+02	1.66E+02	1.66E+02	1.66E+02	1.66E+02	1.66E+02	1.61E+02	1.61E+02	1.20E+02	3.20E+01	6.15E+00
RU-106+RH-106	4.77E+03	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
PD-107	5.85E+02	5.85E+02	5.85E+02	5.85E+02	5.85E+02	5.85E+02	5.85E+02	5.83E+02	5.80E+02	5.57E+02	5.30E+02
AG-110M	9.76E+01	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
CD-113M	2.50E+01	2.10E+00	7.83E+01	1.48E+09	0.	0.	0.	0.	0.	0.	0.
BR-125+Y-125M	3.74E+03	1.00E+02	5.93E+05	0.	0.	0.	0.	0.	0.	0.	0.
SN-126+SB-126	1.13E+02	1.13E+02	1.13E+02	1.13E+02	1.12E+02	1.09E+02	1.05E+02	7.99E+01	5.65E+01	3.54E+00	1.11E-01
I-129	1.61E+01	1.61E+01	1.61E+01	1.61E+01	1.61E+01	1.61E+01	1.61E+01	1.61E+01	1.61E+01	1.58E+01	1.55E+01
CS-134	4.64E+04	2.68E+03	3.12E+06	0.	0.	0.	0.	0.	0.	0.	0.
CS-135	9.33E+01	9.33E+01	9.33E+01	9.33E+01	9.93E+01	9.32E+01	9.93E+01	9.32E+01	9.12E+01	8.31E+01	7.41E+01
CS-137+PA-137	2.36E+06	7.43E+05	4.48E+05	3.97E+01	3.95E+04	0.	0.	0.	0.	0.	0.
CE-144+PR-144	1.55E+03	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
PM-147	9.17E+03	1.66E+02	8.01E+05	0.	0.	0.	0.	0.	0.	0.	0.
SM-151	1.00E+04	9.43E+03	8.74E+03	3.17E+02	5.02E+00	0.	0.	0.	0.	0.	0.
EU-152	4.77E+02	3.22E+01	1.01E+01	6.73E+10	0.	0.	0.	0.	0.	0.	0.
EU-154	1.23E+05	1.01E+04	5.95E+03	1.38E+04	0.	0.	0.	0.	0.	0.	0.
EU-155	1.03E+02	5.05E+07	2.39E+10	0.	0.	0.	0.	0.	0.	0.	0.
DTMFR	2.40E+02	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	4.93E+06	1.43E+06	8.85E+05	6.79E+02	2.85E+02	2.83E+02	2.76E+02	2.59E+02	1.88E+02	4.15E+01	1.09E+01

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.

TABLE A.3.1b. Heat Generation Rates--Once-Through Cycle, Growth Case 1, Watts(A)

Actinides

RADIONUCLIDES (A)	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	7050	2070	500	1000	5000	10000	50000	100000	1000000	
CM-245	2.85E+01	2.46E+01	2.43E+01	2.74E+01	2.43E+01	1.88E+01	1.24E+01	8.91E+00	6.51E+00	0.	0.
CM-246	1.05E+05	1.56E+04	7.18E+03	1.27E+03	0.	0.	0.	0.	0.	0.	0.
CM-243	5.11E+02	1.73E+02	1.12E+02	1.70E+02	3.46E+07	0.	0.	0.	0.	0.	0.
CM-242	1.84E+03	1.47E+03	1.34E+03	2.10E+02	2.19E+01	2.57E+07	0.	0.	0.	0.	0.
AM-243+AM-239	2.96E+03	2.95E+03	2.94E+03	2.84E+03	2.71E+03	1.89E+03	1.20E+03	3.20E+01	3.04E+01	0.	0.
AM-243+AM-242	9.86E+01	7.85E+01	7.16E+01	1.12E+01	1.15E+00	1.37E+00	0.	0.	0.	0.	0.
AM-241	4.93E+05	4.49E+05	8.42E+05	4.47E+05	2.41E+05	3.53E+02	1.73E+01	4.49E+01	4.93E+03	0.	0.
PU-242	2.95E+02	2.95E+02	2.95E+02	2.94E+02	2.94E+02	2.92E+02	2.49E+02	2.49E+02	2.45E+02	1.18E+02	4.73E+01
PU-241	1.28E+04	1.23E+03	4.43E+02	3.63E+02	3.48E+02	2.49E+02	1.44E+02	4.71E+00	4.62E+06	0.	0.
PU-240	1.15E+05	1.14E+05	1.14E+05	1.09E+05	1.04E+05	6.90E+04	4.13E+04	4.43E+02	4.05E+00	0.	0.
PU-239	4.46E+04	4.44E+04	8.44E+04	8.35E+04	8.23E+04	7.37E+04	4.01E+04	2.47E+04	5.00E+03	5.81E+02	3.93E+08
PU-238	7.62E+05	2.46E+05	2.10E+05	9.23E+03	2.25E+02	5.58E+07	0.	0.	0.	0.	0.
PU-236	4.01E+01	4.19E+06	3.24E+08	0.	0.	0.	0.	0.	0.	0.	0.
NP-237+PA-233	4.73E+01	7.89E+01	8.79E+01	1.57E+02	2.45E+02	2.43E+02	2.43E+02	2.20E+02	2.36E+02	2.07E+02	1.76E+02
U-238+Th-234+Pa-234m	9.77E+01	9.77E+01	9.77E+01	9.77E+01	9.77E+01	9.77E+01	9.77E+01	9.78E+01	9.78E+01	9.78E+01	9.78E+01
U-236	4.78E+01	4.79E+01	4.40E+01	4.91E+01	5.45E+01	3.91E+01	4.49E+01	7.48E+01	7.49E+01	7.50E+01	7.39E+01
U-235+Th-231	4.13E+00	4.14E+00	6.13E+00	6.16E+00	6.20E+00	6.48E+00	4.79E+00	4.77E+00	8.68E+00	8.82E+00	8.41E+00
U-234	2.43E+02	3.19E+02	3.30E+02	3.91E+02	3.46E+02	3.43E+02	4.49E+02	4.47E+02	3.22E+02	1.66E+02	1.10E+02
U-233	2.20E+02	4.47E+02	4.33E+02	2.40E+01	6.21E+01	4.41E+00	4.20E+00	4.47E+01	7.46E+01	1.83E+02	1.77E+02
U-232	3.08E+00	1.92E+00	1.54E+00	3.18E+02	2.48E+04	0.	0.	0.	0.	0.	0.
PA-231	7.28E+03	1.44E+02	1.72E+02	7.16E+02	1.42E+01	6.84E+01	1.33E+00	4.29E+00	7.79E+00	9.43E+00	9.42E+00
Th-230	4.72E+02	1.76E+01	2.31E+01	1.47E+00	3.18E+00	1.62E+01	7.18E+01	1.29E+02	2.01E+02	1.92E+02	1.16E+02
Th-229+7 DAUGHTERS	1.03E+04	1.05E+03	1.47E+03	3.47E+02	1.71E+01	5.79E+00	2.19E+01	2.42E+02	4.44E+02	1.26E+03	1.23E+03
Th-228+4 DAUGHTERS	2.00E+01	1.27E+01	1.05E+01	2.10E+01	1.73E+03	1.00E+04	2.18E+04	1.42E+03	2.74E+03	1.41E+02	2.40E+02
Ac-227+7 DAUGHTERS	4.52E+03	6.32E+02	8.76E+02	4.61E+01	9.40E+01	4.47E+00	4.72E+00	4.66E+01	5.09E+01	6.16E+01	6.16E+01
Th-232+2 DAUGHTERS	1.40E+04	1.23E+07	1.74E+07	1.21E+06	2.48E+06	1.44E+05	4.12E+05	1.49E+04	3.42E+04	2.01E+03	4.00E+03
Ra-226+5 DAUGHTERS	1.28E+03	1.46E+02	2.43E+02	7.66E+01	3.24E+00	5.39E+01	1.49E+02	7.20E+02	1.10E+03	1.08E+03	6.55E+02
Pb-210+2 DAUGHTERS	4.05E+05	1.44E+03	2.82E+03	1.61E+01	6.47E+01	1.18E+01	4.00E+01	1.47E+02	2.45E+02	2.32E+02	1.41E+02
TOTAL	1.28E+06	1.32E+06	1.28E+06	6.53E+05	3.41E+05	1.48E+05	1.00E+05	2.48E+04	4.21E+03	3.69E+03	2.90E+03

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.
 B. Th-229, 7 DAUGHTERS ARE RA-225, AC-225, FR-221, AT-217, RI-213, PB-209 AND TL-209 IS 9% OF Th-229 AND SO-213 IS 91% OF Th-229.
 Th-228, 6 DAUGHTERS ARE RA-224, RN-220, PO-216, PB-212, AT-212 AND TL-208 IS 34% OF Th-228 AND PO-218 IS 64% OF Th-228.
 Ac-227, 7 DAUGHTERS ARE Th-227, Pa-223, RN-219, PO-215, PB-211, BI-211 AND TL-207.
 Th-232, 2 DAUGHTERS ARE RA-228 AND AC-228.
 Pa-226, 5 DAUGHTERS ARE RN-222, PO-218, PB-214, BI-214 AND Pn-214.
 Pb-210, 2 DAUGHTERS ARE BI-210 AND PO-210.

NOTE. IN ACCOUNTING FOR THE ACTIVITY IN THIS MANNER, BRANCHING DECAY IN THE CASE OF TL-208 (34%) = PO-212 (64%), AND TL-209 (9%) = PO-211 (91%) WERE COUNTED AS A SINGLE DAUGHTER IN EACH CASE. MINOR BRANCHING (1% OR LESS) WAS IGNORED.

TABLE A.3.2a. Heat Generation Rates--Once-Through Cycle--Growth Case 2, Watts(A)

Fission and Activation Products

MAJOR RADIOISOTOPES	YEAR			GENLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
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M=3	1.14E+02	3.76E+01	1.22E+01	1.43E+09	0.	0.	0.	0.	0.	0.	0.
C=14	4.76E+00	1.14E+01	1.13E+01	1.08E+01	1.02E+01	6.26E+00	4.02E+00	2.72E+02	4.44E+05	0.	0.
MN=4	2.62E+00	4.81E+09	0.	0.	0.	0.	0.	0.	0.	0.	0.
FE=5	4.06E+03	4.43E+00	2.15E+02	0.	0.	0.	0.	0.	0.	0.	0.
CO=60	2.05E+05	7.32E+04	5.26E+02	0.	0.	0.	0.	0.	0.	0.	0.
NI=59	2.83E+00	4.77E+00	6.74E+00	6.70E+00	6.91E+00	6.08E+00	4.21E+00	2.79E+00	2.45E+00	4.92E+02	1.18E+03
NY=7	1.34E+03	2.41E+04	2.07E+03	9.76E+01	2.26E+00	0.	0.	0.	0.	0.	0.
SE=70	2.54E+00	6.20E+00	6.24E+00	6.21E+00	4.14E+00	5.92E+00	4.11E+00	4.47E+00	2.15E+00	3.03E+02	1.07E+04
KR=84	1.08E+05	2.42E+04	7.41E+03	3.43E+08	0.	0.	0.	0.	0.	0.	0.
RA=87	2.18E+04	4.30E+04	5.40E+04	5.30E+04	5.40E+04	4.70E+04	4.40E+04	4.20E+04	3.70E+04	5.30E+04	5.30E+04
SR=90+Y=90	4.23E+06	4.00E+06	3.72E+06	1.68E+02	7.03E+04	0.	0.	0.	0.	0.	0.
ZO=93	3.63E+00	9.26E+00	9.24E+00	9.26E+00	9.26E+00	9.24E+00	9.22E+00	9.25E+00	8.44E+00	7.35E+00	5.43E+00
NR=93M	4.81E+00	2.56E+01	2.47E+01	2.73E+01	2.73E+01	2.73E+01	2.72E+01	2.47E+01	2.41E+01	2.17E+01	1.72E+01
TC=99	4.31E+02	1.04E+03	1.04E+03	1.04E+03	1.04E+03	1.02E+03	1.01E+03	4.22E+02	7.08E+02	2.00E+02	3.44E+01
RI=104+PM=104	1.62E+04	4.62E+04	6.41E+10	0.	0.	0.	0.	0.	0.	0.	0.
PD=107	1.40E+01	3.42E+01	3.42E+01	3.42E+01	3.41E+01	3.41E+01	3.41E+01	3.40E+01	3.74E+01	3.63E+01	3.06E+01
AG=110M	4.56E+00	1.53E+10	0.	0.	0.	0.	0.	0.	0.	0.	0.
CO=113M	9.26E+01	3.94E+01	1.08E+01	2.79E+08	0.	0.	0.	0.	0.	0.	0.
SR=125+YF=125M	1.68E+04	2.20E+01	1.72E+01	0.	0.	0.	0.	0.	0.	0.	0.
SA=126+YF=126	4.05E+02	7.30E+02	7.30E+02	7.26E+02	7.25E+02	7.05E+02	4.41E+02	4.17E+02	3.45E+02	2.29E+01	7.14E+01
I=129	4.24E+01	1.02E+00	1.02E+00	1.02E+00	1.02E+00	1.02E+00	1.02E+00	1.02E+00	1.02E+00	1.00E+00	9.40E+01
CS=134	4.51E+09	4.96E+03	6.92E+02	0.	0.	0.	0.	0.	0.	0.	0.
CS=135	2.51E+00	6.13E+00	6.13E+00	6.13E+00	6.13E+00	6.13E+00	4.12E+00	4.06E+00	4.99E+00	5.46E+00	4.87E+00
CS=137+PA=137	4.64E+06	6.42E+06	4.90E+06	3.63E+02	3.43E+03	0.	0.	0.	0.	0.	0.
CE=140+PR=140	3.14E+03	1.45E+06	0.	0.	0.	0.	0.	0.	0.	0.	0.
PM=147	2.13E+04	2.93E+01	1.08E+01	0.	0.	0.	0.	0.	0.	0.	0.
SM=151	3.44E+04	6.13E+04	5.23E+04	2.06E+03	3.03E+01	0.	0.	0.	0.	0.	0.
EU=152	1.76E+03	4.65E+02	1.78E+02	1.18E+08	0.	0.	0.	0.	0.	0.	0.
EU=154	4.50E+05	2.26E+04	9.49E+04	2.21E+03	0.	0.	0.	0.	0.	0.	0.
EU=155	4.71E+02	3.24E+02	1.94E+05	0.	0.	0.	0.	0.	0.	0.	0.
DTMFR	4.70E+02	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	1.41E+07	1.32E+07	8.18E+06	4.53E+03	1.87E+03	1.79E+03	1.93E+03	1.05E+03	1.16E+03	2.59E+02	6.44E+01

A. VALUES LESS THAN 1.0E+10 HAVE BEEN DESIGNATED AS ZERO.

TABLE A.3.2b. Heat Generation Rates--Once-Through Cycle--Growth Case 2, Watts(A)

Actinides

RADIONUCLIDES (M)	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	
CM-245	1.04E+02	2.52E+02	2.52E+02	2.43E+02	2.43E+02	1.67E+02	1.10E+02	7.43E+00	5.78E+02	0.	0.
CM-244	4.68E+09	2.76E+09	1.28E+05	2.27E+02	1.49E+10	0.	0.	0.	0.	0.	0.
CM-243	1.98E+03	2.11E+03	1.37E+03	2.08E+01	4.11E+06	0.	0.	0.	0.	0.	0.
CM-242	5.52E+03	1.19E+04	1.08E+04	1.70E+03	1.74E+02	2.08E+06	0.	0.	0.	0.	0.
AM-243+MP-239	9.07E+03	2.40E+04	2.39E+04	2.31E+04	2.20E+04	1.53E+04	9.75E+03	2.40E+02	2.80E+00	0.	0.
AM-242M+AM-242	7.18E+02	6.30E+02	5.79E+02	9.09E+01	9.29E+00	1.11E+07	0.	0.	0.	0.	0.
AM-241	1.41E+06	5.28E+06	5.42E+06	2.93E+06	1.42E+06	2.37E+03	1.18E+02	2.48E+00	6.15E+02	0.	0.
PU-242	9.13E+02	2.19E+04	2.19E+03	2.19E+03	2.18E+03	2.17E+03	2.15E+03	2.10E+03	1.22E+03	8.77E+02	3.51E+02
PU-241	4.18E+04	1.48E+04	7.22E+03	3.22E+01	3.09E+01	2.21E+01	1.05E+01	4.07E+03	7.65E+05	0.	0.
PU-240	2.79E+05	6.40E+05	6.59E+05	6.33E+05	6.01E+05	3.99E+05	2.49E+05	1.63E+03	2.34E+01	0.	0.
PU-239	1.80E+05	4.28E+05	4.28E+05	4.23E+05	4.17E+05	3.74E+05	3.26E+05	1.86E+05	2.45E+04	2.96E+01	2.01E+07
PU-238	1.20E+06	2.16E+06	1.85E+06	8.07E+04	1.04E+03	4.51E+06	0.	0.	0.	0.	0.
PU-234	4.27E+00	1.04E+02	5.32E+05	0.	0.	0.	0.	0.	0.	0.	0.
NP-237+PA-233	1.88E+02	4.09E+02	5.41E+02	1.01E+03	1.43E+03	1.58E+03	1.48E+03	1.46E+03	1.54E+03	1.35E+03	1.15E+03
U-238+TH-234+PA-234M	1.95E+02	4.46E+02	4.66E+02	4.66E+02	4.46E+02	4.66E+02	4.46E+02	4.46E+02	4.46E+02	4.66E+02	4.66E+02
U-236	1.18E+02	2.48E+02	2.48E+02	2.95E+02	3.03E+02	3.32E+02	3.02E+02	4.49E+02	4.49E+02	4.49E+02	4.38E+02
U-235+TH-231	9.38E+00	2.40E+01	2.40E+01	2.42E+01	2.40E+01	2.59E+01	2.74E+01	3.04E+01	3.49E+01	3.77E+01	3.77E+01
U-234	5.65E+02	1.48E+03	1.78E+03	2.30E+03	2.45E+03	2.33E+03	2.41E+03	2.11E+03	1.49E+03	9.04E+02	5.52E+02
U-233	4.09E+02	2.01E+01	2.03E+01	1.47E+00	3.07E+00	2.46E+01	4.09E+01	2.45E+02	4.13E+02	1.19E+03	1.15E+03
U-232	1.01E+01	1.75E+01	1.44E+01	2.49E+01	2.45E+03	0.	0.	0.	0.	0.	0.
PA-231	1.88E+02	4.71E+02	7.82E+02	2.87E+01	5.47E+01	2.71E+00	4.42E+00	2.19E+01	3.29E+01	4.03E+01	4.03E+01
TH-230	9.07E+02	7.34E+01	1.03E+00	7.98E+00	1.43E+01	4.46E+01	1.48E+02	7.45E+02	1.18E+03	1.07E+03	5.96E+02
TH-229+ DAUGHTERS	1.54E+04	4.78E+03	7.67E+03	2.04E+01	1.08E+00	3.74E+01	1.02E+02	1.48E+03	3.22E+03	8.24E+03	8.00E+03
TH-228+ DAUGHTERS	4.31E+01	1.16E+02	9.45E+01	1.92E+00	1.47E+02	5.98E+04	1.40E+03	7.43E+03	1.63E+02	8.32E+02	1.46E+01
AC-227+ DAUGHTERS	1.59E+02	2.90E+01	3.40E+01	1.83E+00	3.71E+00	1.77E+01	7.08E+01	1.23E+02	2.15E+02	2.64E+02	2.63E+02
TH-232+ DAUGHTERS	3.42E+08	5.74E+07	8.41E+07	6.95E+06	1.42E+05	8.55E+05	1.46E+04	1.12E+03	2.32E+03	1.19E+02	2.37E+02
PA-226+ DAUGHTERS	2.30E+03	5.10E+02	9.33E+02	3.91E+00	1.41E+01	3.16E+02	4.20E+02	4.42E+03	4.71E+03	6.04E+03	3.35E+03
FR-210+ DAUGHTERS	1.01E+04	4.70E+03	1.00E+02	5.08E+01	3.00E+00	6.40E+01	1.77E+02	9.29E+02	1.04E+03	1.30E+03	7.21E+02
TOTAL	7.59E+06	4.86E+06	8.53E+06	4.10E+06	2.47E+06	7.98E+05	5.43E+05	1.25E+05	4.51E+04	2.22E+04	1.71E+04

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.
 B. TH-229, 7 DAUGHTERS ARE RA-225, AC-229, FR-221, AT-217, BT-213, PB-209 AND TL-204 IS 9% OF TH-229 AND PD-213 IS 9% OF TH-229.
 TH-228, 6 DAUGHTERS ARE RA-224, RN-220, PD-216, PB-212, BT-212 AND TL-208 IS 34% OF TH-228 AND PD-212 IS 64% OF TH-228.
 AC-227, 7 DAUGHTERS ARE TH-227, PA-223, RN-219, PD-215, FR-211, AT-211 AND TL-207.
 TH-232, 2 DAUGHTERS ARE RA-228 AND AC-228.
 PA-226, 5 DAUGHTERS ARE RN-228, PD-218, PB-216, BI-214 AND PH-214.
 PB-210, 2 DAUGHTERS ARE BI-210 AND PD-210.

NOTE. IN ACCOUNTING FOR THE ACTIVITY IN THIS MANNER, BRANCHING DECAY IN THE CASE OF TL-209 (36%) = PD-212 (64%), AND TL-209 (9%) = PD-211 (91%) WERE COUNTED AS A SINGLE DAUGHTER IN EACH CASE. MINOR BRANCHING (1% OR LESS) WAS IGNORED.

TABLE A.3.3a. Heat Generation Rates--Once-Through Cycle--Growth Case 3, Watts(A)

Fission and Activation Products

MAJOR RADIONUCLIDES	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)						
	2000	2050	2070	500	1000	5000	10000	50000	100000	
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H-3	1.88E+02	4.33E+02	1.40E+02	1.65E-08	0.	0.	0.	0.	0.	0.
C-14	7.68E+00	5.89E+01	5.89E+01	5.60E+01	5.27E+01	3.25E+01	1.78E+01	1.01E+01	3.34E+00	0.
MN-54	4.68E+00	4.34E+01	5.22E+08	0.	0.	0.	0.	0.	0.	0.
FE-55	7.62E+03	3.29E+03	1.40E+01	0.	0.	0.	0.	0.	0.	0.
CO-60	3.57E+05	3.30E+05	2.37E+04	0.	0.	0.	0.	0.	0.	0.
NI-63	4.40E+00	3.43E+01	3.42E+01	3.41E+01	3.40E+01	3.24E+01	3.14E+01	2.52E+01	1.44E+01	4.52E-01
NI-63	2.11E+03	1.35E+04	1.14E+04	5.46E+02	1.24E+01	0.	0.	0.	0.	0.
SE-79	3.92E+00	3.13E+01	3.13E+01	3.12E+01	3.10E+01	2.97E+01	2.82E+01	1.44E+01	1.08E+01	1.52E-01
KO-85	1.77E+05	3.49E+05	1.02E+05	5.03E+07	0.	0.	0.	0.	0.	0.
RE-147	3.31E+04	2.46E+04	2.46E+03	2.66E+03	2.46E+03	2.66E+03	2.46E+03	2.46E+03	2.46E+03	2.66E+03
SR-90+Y-90	4.77E+06	4.29E+07	2.42E+07	1.18E+03	5.24E+03	0.	0.	0.	0.	0.
ZR-93	4.81E+00	4.65E+01	4.65E+01	4.65E+01	4.64E+01	4.64E+01	4.43E+01	4.24E+01	4.44E+01	3.69E+01
NR-93	4.82E+00	1.19E+02	1.33E+02	1.37E+02	1.37E+02	1.37E+02	1.37E+02	1.34E+02	1.31E+02	1.09E+02
TC-99	4.45E+02	3.22E+03	5.22E+03	5.21E+03	5.20E+03	5.13E+03	4.95E+03	4.23E+03	3.75E+03	1.00E+03
RU-106+PM-106	2.41E+04	3.42E+03	5.47E+03	0.	0.	0.	0.	0.	0.	0.
PD-107	2.43E+01	1.91E+00	1.91E+00	1.91E+00	1.91E+00	1.91E+00	1.91E+00	1.91E+00	1.90E+00	1.82E+00
AG-110M	4.28E+00	1.17E+00	2.41E+09	0.	0.	0.	0.	0.	0.	0.
CD-113M	1.56E+02	4.11E+02	1.43E+02	2.48E+07	0.	0.	0.	0.	0.	0.
SR-125+TE-125M	3.05E+04	1.31E+04	7.74E+01	0.	0.	0.	0.	0.	0.	0.
SN-128+SB-126	4.63E+02	3.66E+03	3.44E+03	3.45E+03	3.44E+03	3.44E+03	3.42E+03	2.49E+03	1.43E+03	1.15E+02
I-129	4.44E+01	3.12E+00	5.12E+00	5.12E+00	5.12E+00	5.12E+00	4.72E+00	4.11E+00	3.10E+00	5.02E+00
CS-134	4.26E+05	2.02E+05	2.35E+02	0.	0.	0.	0.	0.	0.	0.
CS-135	3.71E+00	3.07E+01	3.07E+01	3.07E+01	3.07E+01	3.04E+01	3.06E+01	3.03E+01	3.00E+01	2.73E+01
CS-137+BA-137	1.04E+07	4.70E+07	2.96E+07	2.51E+03	2.43E+02	0.	0.	0.	0.	0.
CE-140+PR-140	4.55E+03	1.03E+03	1.44E+05	0.	0.	0.	0.	0.	0.	0.
PH-147	4.11E+04	2.06E+04	1.04E+02	0.	0.	0.	0.	0.	0.	0.
SM-151	3.39E+04	3.82E+03	2.92E+05	1.15E+04	2.15E+02	0.	0.	0.	0.	0.
EU-152	2.41E+03	4.44E+03	2.09E+03	1.39E+07	0.	0.	0.	0.	0.	0.
EU-154	3.10E+05	2.11E+04	4.49E+05	2.07E+02	0.	0.	0.	0.	0.	0.
EU-155	1.00E+03	2.96E+02	1.47E+01	0.	0.	0.	0.	0.	0.	0.
OTHER	7.21E+02	1.41E+02	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	2.22E+07	9.34E+07	5.72E+07	2.50E+04	9.81E+03	8.99E+03	4.77E+03	7.28E+03	4.42E+03	1.30E+03

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.

TABLE A.3.3b. Heat Generation Rates--Once-Through Cycle--Growth Case 3, Watts(A)

Actinides

RADIONUCLIDES (A)	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	1000000	
CM-245	1.52E+02	1.28E+03	1.28E+03	1.24E+03	1.19E+03	8.48E+02	4.84E+02	1.05E+01	2.94E+01	0.	0.
CM-246	7.10E+05	2.40E+06	1.12E+06	1.97E+01	9.41E+10	0.	0.	0.	0.	0.	0.
CM-243	2.92E+03	1.43E+04	9.25E+03	1.40E+00	2.77E+05	0.	0.	0.	0.	0.	0.
CM-242	7.77E+03	6.31E+04	5.76E+04	9.03E+03	9.23E+02	1.11E+05	0.	0.	0.	0.	0.
AM-243+MP-239	1.07E+04	1.21E+05	1.21E+05	1.16E+05	1.11E+05	7.75E+04	4.02E+04	7.41E+03	1.41E+01	0.	0.
AM-242+AM-242	4.63E+02	3.37E+03	3.04E+03	4.84E+02	4.04E+01	5.91E+07	0.	0.	0.	0.	0.
AM-241	2.05E+06	2.46E+07	2.46E+07	1.50E+07	6.77E+06	1.21E+04	4.04E+02	2.47E+01	3.13E+01	0.	0.
PU-242	1.36E+03	1.10E+04	1.10E+04	1.10E+04	1.10E+04	1.09E+04	1.08E+04	1.07E+04	9.16E+03	4.91E+03	1.77E+03
PU-241	6.83E+04	1.82E+05	7.14E+04	1.64E+00	1.47E+00	1.12E+00	7.49E+01	2.24E+02	3.49E+04	0.	0.
PU-240	4.35E+05	3.31E+06	3.30E+06	3.17E+06	3.01E+06	2.00E+06	1.20E+06	1.08E+04	1.18E+02	0.	0.
PU-239	2.86E+04	2.15E+06	2.14E+06	2.12E+06	2.09E+06	1.88E+04	1.43E+04	5.20E+05	1.28E+05	1.40E+00	1.01E+06
PU-238	1.78E+06	1.20E+07	1.03E+07	4.49E+05	1.07E+04	2.40E+05	0.	0.	0.	0.	0.
PU-236	1.10E+01	4.71E+00	3.43E+02	0.	0.	0.	0.	0.	0.	0.	0.
NP-237+PA-233	2.81E+02	2.46E+03	2.62E+03	4.94E+03	6.66E+03	7.96E+03	7.06E+03	7.86E+03	7.73E+03	4.79E+03	5.78E+03
U-234+Th-234+Pa-234m	3.13E+02	2.33E+03	2.33E+03	2.33E+03	2.33E+03	2.33E+03	2.33E+03	2.33E+03	2.33E+03	2.33E+03	2.33E+03
U-236	1.79E+02	1.44E+03	1.44E+03	1.48E+03	1.47E+03	1.77E+03	1.06E+03	2.25E+03	2.25E+03	2.23E+03	2.20E+03
U-235+Th-231	1.52E+01	1.20E+02	1.21E+02	1.21E+02	1.22E+02	1.29E+02	1.27E+02	1.22E+02	1.25E+02	1.89E+02	1.49E+02
U-234	8.52E+02	4.03E+03	4.59E+03	1.15E+04	1.14E+04	1.17E+04	1.16E+04	1.16E+04	9.08E+03	4.55E+03	2.76E+03
U-233	7.57E+02	4.79E+01	1.08E+00	6.98E+00	1.05E+01	1.43E+02	3.01E+02	1.03E+03	2.49E+03	5.99E+03	5.81E+03
U-232	1.50E+01	1.00E+02	8.24E+01	1.66E+00	1.25E+02	0.	0.	0.	0.	0.	0.
PA-231	2.67E+02	3.00E+01	3.46E+01	1.34E+00	2.00E+00	1.35E+01	2.46E+01	1.10E+02	1.45E+02	2.02E+02	2.02E+02
Th-230	1.27E+01	2.74E+00	4.20E+00	3.79E+01	2.00E+01	4.78E+02	9.44E+02	3.84E+03	4.93E+03	5.37E+03	2.98E+03
Th-229+7 DAUGHTERS	1.44E+04	1.71E+02	2.99E+02	9.32E+01	5.21E+00	1.86E+02	7.13E+02	7.92E+03	1.42E+04	4.15E+04	4.02E+04
Th-228+4 DAUGHTERS	9.15E+01	6.40E+02	5.07E+02	1.10E+01	2.04E+02	2.99E+03	4.01E+03	1.03E+02	4.15E+02	4.17E+01	8.31E+01
Ac-227+7 DAUGHTERS	1.86E+02	1.14E+00	1.43E+00	8.75E+00	1.03E+01	8.82E+01	1.74E+02	7.17E+02	1.08E+03	1.32E+03	1.32E+03
Th-232+3 DAUGHTERS	4.06E+04	2.09E+06	3.40E+06	3.33E+05	7.43E+05	4.27E+04	9.31E+04	5.11E+03	1.16E+02	5.96E+02	1.19E+01
Ra-226+4 DAUGHTERS	3.00E+03	1.40E+01	3.26E+01	1.78E+01	8.49E+01	1.97E+03	4.13E+03	2.17E+04	3.37E+04	3.03E+04	1.68E+04
Pb-210+2 DAUGHTERS	1.24E+04	1.33E+02	3.96E+02	3.62E+00	1.04E+01	3.38E+02	8.27E+02	4.46E+03	7.24E+03	6.51E+03	3.61E+03
TOTAL	4.38E+06	4.49E+07	4.40E+07	2.10E+07	1.20E+07	4.00E+06	2.02E+06	4.25E+05	2.26E+05	1.12E+05	8.49E+04

A. VALUES LESS THAN 1.0E+10 HAVE BEEN DESIGNATED AS ZERO.
 B. Th-229, 7 DAUGHTERS ARE Ra-225, Ac-225, Fr-221, At-217, Rf-213, Pb-209 AND Tl-209 IS 9% OF Th-229 AND Po-213 IS 9% OF Th-229.
 Th-228, 6 DAUGHTERS ARE Ra-228, Rn-220, Po-216, Pb-212, Bi-212 AND Tl-208 IS 3% OF Th-228 AND Po-214 IS 6% OF Th-228.
 Ac-227, 7 DAUGHTERS ARE Th-227, Ra-223, Rn-219, Po-215, Pb-211, Bi-211 AND Tl-207.
 Th-232, 3 DAUGHTERS ARE Ra-228, Ac-228.
 Ra-226, 5 DAUGHTERS ARE Rn-222, Po-214, Pb-210, Bi-210 AND Po-214.
 Pb-210, 2 DAUGHTERS ARE Bi-210 AND Po-210.

NOTE: IN ACCOUNTING FOR THE ACTIVITY IN THIS MANNER, BRANCHING DECAY IN THE CASE OF Tl-208 (3%) = Po-212 (6%), AND Tl-209 (9%) = Po-211 (9%) WERE COUNTED AS A SINGLE DAUGHTER IN EACH CASE. MINOR BRANCHING (1% OR LESS) WAS IGNORED.

TABLE A.3.4a. Heat Generation Rates--Once-Through Cycle--Growth Case 4, Watts(A)

Fission and Activation Products

MAJOR RADIOISOTOPES	YEAR			DEGRADATION TIME (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----
H-3	1.41E+00	4.33E+02	2.71E+02	3.18E-08	0.	0.	0.	0.	0.	0.	0.
C-14	1.09E+01	7.40E+01	7.76E+01	7.41E+01	4.07E+01	4.30E+01	2.33E+01	1.27E+01	4.42E+00	0.	0.
MN-54	2.89E+07	7.49E+00	4.16E+07	0.	0.	0.	0.	0.	0.	0.	0.
FE-55	7.62E+00	1.60E+04	7.76E+01	0.	0.	0.	0.	0.	0.	0.	0.
CO-60	0.55E+02	1.01E+04	7.23E+04	0.	0.	0.	0.	0.	0.	0.	0.
NI-63	4.68E+02	4.92E+01	4.92E+01	4.50E+01	4.44E+01	4.33E+01	4.14E+01	2.43E+01	1.00E+01	5.95E+01	7.45E+03
NI-67	2.92E+01	1.46E+00	1.60E+04	7.52E+02	1.74E+01	0.	0.	0.	0.	0.	0.
SE-79	4.10E+02	4.11E+01	4.11E+01	4.09E+01	4.47E+01	3.90E+01	7.70E+01	2.02E+01	1.42E+01	2.00E+01	4.64E+04
KR-85	1.22E+03	7.46E+05	2.07E+05	1.01E+06	0.	0.	0.	0.	0.	0.	0.
Rb-87	4.17E+06	1.49E+03	3.49E+03	3.49E+03	3.49E+03	3.49E+03	7.49E+03	7.49E+03	3.49E+03	3.49E+03	3.49E+03
SR-90+Y-90	1.12E+09	6.77E+07	4.01E+07	1.41E+03	4.42E+03	0.	0.	0.	0.	0.	0.
ZO-93	4.07E+02	6.10E+01	6.10E+01	6.10E+01	6.10E+01	6.09E+01	6.07E+01	5.06E+01	5.02E+01	4.84E+01	3.44E+01
MO-93M	2.03E+01	1.46E+02	1.68E+02	1.80E+02	1.80E+02	1.80E+02	1.79E+02	1.76E+02	1.72E+02	1.43E+02	1.14E+02
TC-99	1.03E+01	6.45E+03	6.45E+03	6.44E+03	6.43E+03	6.74E+03	4.43E+03	4.41E+03	4.03E+03	1.32E+03	2.53E+02
RU-106+PM-106	4.50E+03	4.38E+04	4.38E+02	0.	0.	0.	0.	0.	0.	0.	0.
PO-107	7.75E+03	2.92E+00	2.92E+00	2.52E+00	2.42E+00	2.52E+00	2.42E+00	2.41E+00	2.40E+00	2.40E+00	2.78E+00
AG-110M	1.19E+04	1.74E+01	2.64E+08	0.	0.	0.	0.	0.	0.	0.	0.
CO-113M	1.04E+00	7.40E+02	2.43E+02	5.33E+07	0.	0.	0.	0.	0.	0.	0.
SB-125+TE-125M	1.25E+01	6.91E+04	3.45E+02	0.	0.	0.	0.	0.	0.	0.	0.
SM-126+SR-126	7.13E+00	4.42E+03	4.42E+03	4.41E+03	4.79E+03	4.46E+03	4.40E+03	4.41E+03	2.41E+03	1.51E+02	4.74E+00
I-129	1.00E+02	4.73E+00	6.73E+00	4.73E+00	4.73E+00	6.73E+00	4.73E+00	4.72E+00	4.70E+00	6.40E+00	6.46E+00
CS-134	4.67E+01	1.24E+06	1.44E+03	0.	0.	0.	0.	0.	0.	0.	0.
CS-135	4.07E+02	4.01E+01	4.01E+01	4.01E+01	4.01E+01	4.00E+01	4.00E+01	4.00E+01	3.92E+01	3.57E+01	3.18E+01
CS-137+BA-137	1.22E+05	7.13E+07	4.49E+07	3.81E+03	3.49E+02	0.	0.	0.	0.	0.	0.
CE-144+PR-144	4.12E+05	7.48E+03	1.47E+04	0.	0.	0.	0.	0.	0.	0.	0.
PM-147	1.88E+01	4.91E+04	4.40E+02	0.	0.	0.	0.	0.	0.	0.	0.
SM-151	4.52E+02	4.72E+05	4.02E+05	1.59E+04	2.46E+02	0.	0.	0.	0.	0.	0.
EU-152	1.89E+01	1.27E+04	4.01E+03	2.66E+07	0.	0.	0.	0.	0.	0.	0.
EU-154	4.30E+03	3.72E+06	1.46E+06	3.64E+02	0.	0.	0.	0.	0.	0.	0.
EU-155	4.53E+02	1.93E+03	4.17E+01	0.	0.	0.	0.	0.	0.	0.	0.
OTHFR	0.	1.15E+01	1.03E+10	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	2.42E+05	1.44E+08	8.73E+07	3.43E+04	1.24E+04	1.18E+04	1.18E+04	4.56E+03	7.65E+03	1.71E+03	4.51E+02

A. VALUES LESS THAN 1.0E+10 HAVE BEEN DESIGNATED AS ZERO.

TABLE A.3.4b. Heat Generation Rates--Once Through Cycle--Growth Case 4, Watts(A)

Actinides

RADIONUCLIDES (g)	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
CM-245	1.50E+00	1.68E+03	1.64E+03	1.62E+03	1.45E+03	1.11E+03	7.40E+02	2.43E+01	3.43E+01	0.	0.
CM-244	4.03E+03	4.07E+06	1.69E+06	3.34E+01	1.61E+09	0.	0.	0.	0.	0.	0.
CM-243	2.38E+01	2.14E+04	1.39E+04	2.10E+00	4.15E+03	0.	0.	0.	0.	0.	0.
CM-242	5.24E+01	4.49E+04	7.75E+04	1.22E+04	1.24E+03	1.49E+03	0.	0.	0.	0.	0.
AM-243+MP-230	1.66E+02	1.49E+05	1.49E+05	1.53E+05	1.06E+05	1.02E+05	4.06E+04	1.72E+03	1.46E+01	0.	0.
AM-242M+AM-242	4.19E+00	4.84E+03	4.15E+03	6.51E+02	6.48E+01	7.95E+07	0.	0.	0.	0.	0.
AM-241	4.47E+04	7.02E+07	3.47E+07	2.00E+07	9.01E+06	1.62E+04	7.44E+02	2.72E+01	4.10E+01	0.	0.
PU-242	1.76E+01	1.45E+04	1.45E+04	1.44E+04	1.44E+04	1.43E+04	1.42E+04	1.42E+04	1.20E+04	5.74E+03	2.32E+03
PU-241	4.68E+02	7.32E+04	1.30E+05	2.14E+00	2.06E+00	1.47E+00	4.47E+01	7.44E+02	4.10E+04	0.	0.
PU-240	7.87E+03	4.37E+06	4.37E+06	4.20E+06	3.49E+06	2.64E+06	1.48E+06	2.42E+04	1.95E+02	0.	0.
PU-239	4.05E+04	2.44E+06	2.44E+06	2.81E+06	2.77E+06	2.48E+06	2.16E+06	7.61E+05	1.70E+05	1.97E+00	1.33E+06
PL-238	1.94E+04	1.65E+07	1.41E+07	6.15E+05	1.47E+04	3.23E+05	0.	0.	0.	0.	0.
PL-236	4.00E+03	2.41E+01	1.86E+01	0.	0.	0.	0.	0.	0.	0.	0.
NP-237+PA-233	4.10E+00	3.18E+03	3.38E+03	6.42E+03	4.74E+03	1.05E+04	1.45E+04	1.44E+04	1.42E+04	4.95E+03	7.61E+03
U-238+Th-234+Pa-234m	4.01E+00	3.04E+04	3.04E+04	3.08E+03	3.04E+03	3.08E+04	4.04E+03	3.04E+03	4.04E+03	3.08E+03	3.08E+03
U-236	2.65E+00	1.49E+03	1.49E+03	1.93E+03	1.49E+03	2.32E+03	2.48E+03	2.66E+03	2.46E+03	2.93E+03	2.49E+03
U-235+Th-231	3.52E+01	1.57E+02	1.47E+02	1.58E+02	1.49E+02	1.48E+02	1.79E+02	2.55E+02	2.42E+02	2.47E+02	2.47E+02
U-234	1.45E+01	1.02E+04	1.10E+04	1.48E+04	1.44E+04	1.52E+04	1.41E+04	1.44E+04	1.24E+04	5.96E+03	3.63E+03
U-233	1.35E+03	1.04E+00	1.34E+00	8.84E+00	2.44E+01	1.48E+02	4.06E+02	1.44E+03	4.00E+03	7.90E+03	7.45E+03
U-232	1.60E+01	1.38E+02	1.14E+02	2.30E+00	1.46E+02	0.	0.	0.	0.	0.	0.
PA-231	4.51E+04	7.71E+01	4.44E+01	1.76E+00	3.41E+00	1.75E+01	4.47E+01	1.43E+02	2.16E+02	2.65E+02	2.65E+02
TH-230	3.48E+03	3.18E+00	4.96E+00	4.77E+01	1.16E+02	6.20E+02	1.23E+03	4.00E+03	7.73E+03	7.02E+03	3.42E+03
TH-229+7 DAUGHTERS	4.67E+08	1.69E+02	3.27E+02	1.15E+00	6.72E+00	2.43E+02	4.44E+02	1.04E+04	2.14E+04	5.46E+04	5.30E+04
TH-228+6 DAUGHTERS	4.19E+01	9.03E+02	7.88E+02	1.52E+01	1.34E+01	3.40E+03	4.44E+03	4.76E+02	1.07E+01	5.48E+01	1.09E+00
AC-227+7 DAUGHTERS	4.38E+06	1.24E+00	1.92E+00	1.11E+01	2.46E+01	1.15E+02	4.27E+02	4.46E+02	1.41E+03	1.73E+03	1.73E+03
TH-227+2 DAUGHTERS	2.41E+09	1.90E+06	3.80E+06	4.20E+05	9.74E+05	5.58E+04	1.22E+03	7.47E+03	1.43E+02	7.84E+02	1.56E+01
RA-226+4 DAUGHTERS	1.24E+04	1.71E+01	3.66E+01	2.18E+01	1.12E+02	2.04E+03	4.47E+03	2.42E+04	4.39E+04	3.95E+04	2.21E+04
PB-214+2 DAUGHTERS	4.87E+06	1.34E+02	3.54E+02	4.38E+00	2.42E+01	4.38E+02	1.15E+03	6.07E+03	4.43E+03	8.50E+03	4.74E+03
TOTAL	4.32E+04	5.46E+07	5.44E+07	2.74E+07	1.40E+07	5.29E+06	3.47E+06	4.55E+05	2.48E+05	1.47E+05	1.13E+05

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.
 B. TH-229, 7 DAUGHTERS ARE RA-223, AC-225, FR-221, AT-217, RI-213, PB-209 AND TL-204 IS 9% OF TH-229 AND PO-213 IS 91% OF TH-229.
 TH-228, 6 DAUGHTERS ARE RA-224, RN-220, PO-216, PB-212, RI-212 AND TL-208 IS 34% OF TH-228 AND PN-212 IS 64% OF TH-228.
 AC-227, 7 DAUGHTERS ARE TH-227, RA-223, RN-219, PO-213, PA-211, BI-211 AND TL-207.
 TH-222, 2 DAUGHTERS ARE RA-228 AND AC-228.
 PA-226, 5 DAUGHTERS ARE RN-222, PO-218, PB-214, BI-214 AND PN-214.
 PB-210, 2 DAUGHTERS ARE BI-210 AND PO-210.

NOTE. IN ACCOUNTING FOR THE ACTIVITY IN THIS MANNER, BRANCHING DECAY IN THE CASE OF PL-208 (3%) = PO-212 (64%), AND TL-209 (9%) = PO-211 (91%) WERE COUNTED AS A SINGLE DAUGHTER IN EACH CASE. MINOR BRANCHING (1% OR LESS) WAS IGNORED.

TABLE A.3.5a. Heat Generation Rates--Once-Through Cycle--Growth Case 5, Watts(A)

Fission and Activation Products

MAJOR RADIONUCLIDES	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)						
	2000	2050	2070	500	1000	5000	10000	50000	100000	1000000
H-3	1.01E+00	1.35E+03	0.96E+02	5.13E-08	0.	0.	0.	0.	0.	0.
C-10	1.09E+01	1.06E+02	1.06E+02	1.01E+02	9.41E+01	5.86E+01	4.20E+01	2.54E+01	6.03E+00	0.
MN-54	2.89E+07	1.42E+01	7.93E+07	0.	0.	0.	0.	0.	0.	0.
FE-55	4.62E+00	2.95E+00	1.43E+02	0.	0.	0.	0.	0.	0.	0.
CO-60	9.55E+02	1.79E+04	1.28E+05	0.	0.	0.	0.	0.	0.	0.
NI-63	6.68E+02	6.10E+01	6.14E+01	6.12E+01	6.10E+01	5.89E+01	4.44E+01	4.09E+01	2.59E+01	8.10E+01
NI-64	2.92E+01	2.40E+04	2.24E+04	1.05E+03	2.04E+01	0.	0.	0.	0.	0.
SE-79	6.10E+02	4.99E+01	5.44E+01	5.56E+01	5.43E+01	5.30E+01	4.03E+01	3.28E+01	1.03E+01	2.71E+01
KR-84	1.22E+03	1.22E+04	3.98E+05	1.66E+06	0.	0.	0.	0.	0.	0.
RA-227	4.17E+06	4.73E+03	4.73E+03	4.73E+03	4.73E+03	4.73E+03	4.73E+03	4.73E+03	4.73E+03	4.73E+03
SR-90+Y-90	1.12E+05	9.72E+07	5.91E+17	2.67E+05	1.19E+02	0.	0.	0.	0.	0.
ZR-93	9.07E+02	9.24E+01	8.24E+01	9.24E+01	8.24E+01	8.24E+01	8.24E+01	8.24E+01	8.24E+01	8.24E+01
MG-131	2.03E+01	1.90E+02	2.25E+02	2.44E+02	2.65E+02	2.44E+02	2.04E+02	2.49E+02	2.34E+02	1.94E+02
TC-99	1.03E+01	9.31E+03	9.31E+03	9.30E+03	9.28E+03	9.18E+03	9.01E+03	7.00E+03	4.70E+03	1.79E+03
RU-106+RH-106	9.50E+03	8.35E+04	8.99E+02	0.	0.	0.	0.	0.	0.	0.
PD-107	4.75E+03	4.43E+00	3.04E+00	3.43E+00	3.03E+00	3.43E+00	4.03E+00	4.02E+00	4.00E+00	3.27E+00
AG-110M	1.19E+08	2.50E+01	5.16E+08	0.	0.	0.	0.	0.	0.	0.
CD-113M	1.00E+00	1.22E+03	9.42E+02	9.53E-07	0.	0.	0.	0.	0.	0.
SA-125+TE-125M	1.25E+01	1.21E+05	7.14E+02	0.	0.	0.	0.	0.	0.	0.
SA-126+FR-126	7.13E+00	4.56E+03	6.96E+03	6.54E+03	6.42E+03	6.34E+03	4.12E+03	4.44E+03	3.28E+03	2.04E+02
I-129	1.00E+02	9.15E+00	9.15E+00	9.15E+00	9.15E+00	9.15E+00	9.10E+00	9.13E+00	9.11E+00	8.96E+00
CS-134	4.67E+01	2.34E+06	2.72E+03	0.	0.	0.	0.	0.	0.	0.
CS-135	4.07E+02	5.04E+01	5.04E+01	5.04E+01	5.04E+01	5.03E+01	5.03E+01	5.08E+01	5.01E+01	4.85E+01
CS-137+SA-137	1.22E+04	1.05E+04	6.42E+07	5.62E+03	5.04E+02	0.	0.	0.	0.	0.
CE-144+R-144	6.12E+05	1.44E+04	2.63E+04	0.	0.	0.	0.	0.	0.	0.
PM-147	1.88E+01	1.43E+05	8.22E+02	0.	0.	0.	0.	0.	0.	0.
SM-151	8.52E+02	6.49E+04	5.42E+05	2.22E+04	4.14E+02	0.	0.	0.	0.	0.
EU-152	1.99E+01	2.05E+04	6.07E+03	6.29E+07	0.	0.	0.	0.	0.	0.
EU-154	4.30E+03	5.84E+06	2.88E+06	5.71E+02	0.	0.	0.	0.	0.	0.
EU-155	4.53E+02	3.66E+03	1.74E+00	0.	0.	0.	0.	0.	0.	0.
DTMFR	0.	2.25E+01	2.07E+10	0.	0.	0.	0.	0.	0.	0.
TOTAL	2.02E+04	2.15E+04	1.29E+08	4.80E+04	1.48E+04	1.61E+04	1.47E+04	1.40E+04	1.00E+04	2.32E+03

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.

TABLE A.3.5b. Heat Generation Rates--Once Through Cycle--Growth Case 5, Watts(A)

Actinides

RADIONUCLIDES (A)	YEAR			GEOLGIC TIME (YEARS BEYOND 1995)							
	2040	2090	2070	500	1000	5000	10000	50000	100000	500000	1000000
CM=245	1.50E+00	2.29E+03	2.27E+03	2.21E+03	2.12E+03	1.51E+03	9.06E+02	7.88E+01	5.25E+01	0.	0.
CM=244	4.43E+03	6.32E+06	2.94E+06	5.19E+01	2.81E+09	0.	0.	0.	0.	0.	0.
CM=243	2.36E+01	3.15E+04	2.04E+04	3.09E+00	4.72E+05	0.	0.	0.	0.	0.	0.
CM=242	5.24E+01	1.18E+04	1.07E+05	1.68E+04	1.72E+03	2.06E+05	0.	0.	0.	0.	0.
AM=243+AM=233	1.66E+02	2.16E+05	2.14E+05	2.08E+05	1.09E+05	1.39E+05	4.20E+04	2.45E+03	2.43E+01	0.	0.
AM=242+AM=242	5.19E+00	6.29E+03	5.74E+03	9.01E+02	9.22E+01	1.10E+06	0.	0.	0.	0.	0.
AM=241	0.07E+04	3.94E+07	4.48E+07	2.75E+07	1.29E+07	2.21E+04	1.07E+03	7.70E+01	4.49E+01	0.	0.
PU=242	1.76E+01	1.97E+04	1.97E+04	1.97E+04	1.97E+04	1.95E+04	1.03E+04	1.20E+04	1.44E+04	7.89E+03	3.16E+03
PI=241	5.88E+02	5.25E+04	2.04E+05	2.92E+00	2.01E+00	2.01E+00	1.32E+00	0.40E+02	4.95E+04	0.	0.
PU=240	7.87E+03	5.95E+06	5.95E+06	5.71E+06	5.43E+06	3.60E+06	2.76E+06	4.74E+04	2.12E+02	0.	0.
PU=230	4.95E+03	3.86E+06	3.45E+06	3.81E+06	3.76E+06	3.37E+06	2.99E+06	4.22E+05	2.30E+05	2.67E+00	1.41E+06
PU=234	1.94E+04	2.31E+07	1.94E+07	8.81E+05	2.03E+04	4.47E+05	0.	0.	0.	0.	0.
PI=234	4.00E+03	4.92E+01	3.49E+01	0.	0.	0.	0.	0.	0.	0.	0.
NP=237+PA=233	4.10E+00	4.28E+03	4.54E+03	8.67E+03	1.19E+04	1.43E+04	1.03E+04	1.21E+04	1.49E+04	1.22E+04	1.04E+04
U=238+Th=234+Pa=234	4.91E+00	4.17E+03	4.17E+03	4.17E+03	4.17E+03	4.17E+03	4.17E+03	4.17E+03	4.17E+03	4.17E+03	4.17E+03
U=236	2.25E+00	2.86E+03	2.57E+03	2.62E+03	2.70E+03	3.14E+03	4.00E+03	4.22E+03	4.02E+03	3.94E+03	3.92E+03
U=235+Th=231	3.42E+01	2.10E+02	2.10E+02	2.11E+02	2.13E+02	2.28E+02	2.40E+02	3.03E+02	3.26E+02	3.33E+02	3.33E+02
U=234	1.45E+01	1.35E+04	1.05E+04	2.00E+04	2.08E+04	2.08E+04	2.04E+04	1.86E+04	1.67E+04	4.06E+03	4.92E+03
U=233	1.35E+03	1.40E+00	1.75E+00	1.18E+01	3.03E+01	2.55E+02	4.49E+02	2.47E+03	4.62E+03	1.08E+04	1.04E+04
U=232	1.60E+01	1.90E+02	1.61E+02	3.24E+00	2.43E+02	0.	0.	0.	0.	0.	0.
PA=231	4.91E+04	4.05E+01	5.42E+01	2.33E+00	4.02E+00	2.35E+01	4.45E+01	1.03E+02	2.90E+02	3.56E+02	3.56E+02
Th=230	4.08E+03	3.08E+00	6.24E+00	4.35E+01	1.46E+02	8.37E+02	1.06E+03	4.76E+03	1.04E+04	9.50E+03	9.31E+03
Th=229+ DAUGHTERS	4.87E+04	2.00E+02	4.05E+02	1.52E+04	9.03E+00	5.30E+02	1.28E+03	1.22E+04	2.91E+04	7.44E+04	7.22E+04
Th=228+ DAUGHTERS	9.19E+01	1.27E+03	1.07E+03	2.14E+01	1.75E+01	5.28E+03	1.16E+02	7.10E+02	1.45E+01	7.44E+01	1.08E+00
AC=227+ DAUGHTERS	9.38E+06	1.51E+00	2.06E+00	1.47E+01	3.15E+01	1.53E+02	4.00E+02	1.36E+03	1.40E+03	2.33E+03	2.33E+03
Th=226+ DAUGHTERS	2.91E+09	2.24E+06	4.77E+06	5.62E+05	1.42E+04	7.55E+04	1.45E+03	1.00E+02	2.08E+02	1.06E+01	2.12E+01
RA=226+ DAUGHTERS	1.24E+04	1.94E+01	4.37E+01	2.86E+01	1.40E+02	2.75E+03	7.95E+03	4.11E+04	5.93E+04	5.35E+04	2.99E+04
PB=210+ DAUGHTERS	4.87E+06	1.92E+02	4.10E+02	5.72E+00	3.23E+01	5.91E+02	1.46E+03	8.20E+03	1.27E+04	1.15E+04	6.02E+03
TOTAL	4.32E+04	7.96E+07	7.09E+07	3.81E+07	2.14E+07	7.20E+06	4.26E+06	1.72E+06	4.04E+05	1.99E+05	1.54E+05

A. VALUES LESS THAN 1.0E+10 HAVE BEEN DESIGNATED AS ZERO.
B. TH=229, 7 DAUGHTERS ARE RA=225, AC=225, FR=221, AT=217, RI=213, PB=209 AND TL=200 IS 9% OF TH=229 AND PO=213 IS 91% OF TH=229.
TH=228, 6 DAUGHTERS ARE RA=224, RN=220, PO=216, PB=212, RI=212 AND TL=208 IS 54% OF TH=228 AND PO=212 IS 62% OF TH=228.
AC=227, 7 DAUGHTERS ARE TH=227, RA=223, RN=219, PO=215, PR=211, RI=211 AND TL=207.
TH=226, 2 DAUGHTERS ARE RA=226 AND AC=226.
PA=226, 5 DAUGHTERS ARE RN=222, PN=218, PB=210, RI=210 AND PR=210.
PB=210, 2 DAUGHTERS ARE RI=210 AND PO=210.

NOTE. IN ACCOUNTING FOR THE ACTIVITY IN THIS MANNER, BRANCHING DECAY IN THE CASE OF TL=208 (34%) - PO=212 (64%), AND TL=209 (4%) - PO=211 (41%) WERE COUNTED AS A SINGLE DAUGHTER IN EACH CASE. MINOR BRANCHING (1% OR LESS) WAS IGNORED.

TABLE A.3.6a. Heat Generation Rates--Reprocessing Cycle--Growth Case 3--1990 Reprocessing Startup, Watts(A)

Fission and Activation Products

MAJOR RADIOISOTOPES	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	300	1000	5000	10000	50000	100000	500000	1000000
1-3	2.65E+01	4.99E+01	2.27E+01	2.66E-09	0.	0.	0.	0.	0.	0.	0.
2-10	8.09E+00	9.62E+01	5.60E+01	5.36E+01	5.72E+01	3.10E+01	1.49E+01	1.74E+01	3.19E+04	0.	0.
4N-50	3.42E+00	1.03E+00	5.73E-08	0.	0.	0.	0.	0.	0.	0.	0.
7E-85	6.72E+03	2.73E+03	1.33E+01	0.	0.	0.	0.	0.	0.	0.	0.
20-60	3.29E+05	3.03E+05	2.14E+04	0.	0.	0.	0.	0.	0.	0.	0.
41-40	4.21E+00	2.98E+01	2.98E+01	2.97E+01	2.05E+01	2.85E+01	2.73E+01	1.03E+01	1.25E+01	3.92E+01	5.17E-03
41-63	2.02E+03	1.17E+04	1.00E+04	4.71E+02	1.09E+01	0.	0.	0.	0.	0.	0.
3E-79	1.23E+00	2.98E+01	2.97E+01	2.96E+01	2.05E+01	2.82E+01	2.48E+01	1.73E+01	1.03E+01	1.40E+01	7.01E-06
1R-85	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
2B-87	1.04E-04	2.36E-03	2.36E-03	2.36E-03	2.36E-03	2.36E-03	2.36E-03	2.36E-03	2.36E-03	2.36E-03	2.36E-03
3R-00+Y-90	2.66E+06	3.74E+07	2.28E+07	1.03E+03	4.46E-03	0.	0.	0.	0.	0.	0.
2R-03	2.16E+00	4.80E+01	4.60E+01	4.60E+01	4.80E+01	4.59E+01	4.98E+01	4.20E+01	4.80E+01	3.65E+01	2.40E+01
4B-03M	4.31E+00	1.18E+02	1.29E+02	1.36E+02	1.46E+02	1.36E+02	1.45E+02	1.33E+02	1.30E+02	1.08E+02	8.57E+01
1C-09	2.07E+02	9.17E+03	5.17E+03	5.16E+03	5.15E+03	5.09E+03	4.80E+03	4.49E+03	3.72E+03	9.90E+02	1.91E+02
1U-106+PH-106	6.68E+01	6.28E+03	6.44E-03	0.	0.	0.	0.	0.	0.	0.	0.
70-107	7.67E+02	2.39E+00	2.39E+00	2.39E+00	2.39E+00	2.39E+00	2.39E+00	2.39E+00	2.39E+00	2.27E+00	2.16E+00
16-110M	4.90E+03	1.70E+00	3.09E-09	0.	0.	0.	0.	0.	0.	0.	0.
20-113M	8.78E+01	8.34E+02	3.10E+02	5.93E+07	0.	0.	0.	0.	0.	0.	0.
18-125+Y-125M	1.85E+03	1.99E+04	9.34E+01	0.	0.	0.	0.	0.	0.	0.	0.
1N-126+ZB-126	1.46E+02	4.27E+03	4.27E+03	4.26E+03	4.24E+03	4.13E+03	3.99E+03	3.82E+03	2.14E+03	1.30E+02	4.20E+00
1-129	6.13E+01	5.46E+00	5.46E+00	5.46E+00	5.46E+00	5.46E+00	5.45E+00	5.45E+00	5.43E+00	5.35E+00	5.24E+00
18-130	1.85E+04	2.03E+04	2.35E+02	0.	0.	0.	0.	0.	0.	0.	0.
18-135	1.13E+00	3.86E+01	3.86E+01	3.86E+01	3.86E+01	3.85E+01	3.85E+01	3.81E+01	3.77E+01	3.40E+01	3.06E+01
18-137+PA-137	2.80E+06	4.89E+07	2.95E+07	2.51E+03	2.43E+02	0.	0.	0.	0.	0.	0.
1E-140+PR-144	3.70E+00	1.01E+03	1.84E-05	0.	0.	0.	0.	0.	0.	0.	0.
1M-147	2.19E+03	2.00E+04	1.01E+02	0.	0.	0.	0.	0.	0.	0.	0.
1M-151	1.65E+04	3.45E+05	3.11E+05	1.23E+04	2.29E+02	0.	0.	0.	0.	0.	0.
1U-152	6.21E+02	1.15E+04	3.44E+03	2.41E+07	0.	0.	0.	0.	0.	0.	0.
1U-154	1.68E+05	2.55E+06	1.07E+06	2.50E+02	0.	0.	0.	0.	0.	0.	0.
1U-155	2.13E+01	3.52E+02	1.67E+01	0.	0.	0.	0.	0.	0.	0.	0.
1YFR	1.89E+02	1.31E+02	0.	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	6.07E+06	8.78E+07	5.34E+07	2.60E+04	9.08E+03	9.43E+03	9.29E+03	7.27E+03	6.10E+03	1.31E+03	3.48E+02

VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.

TABLE A.3.6b. Heat Generation Rates, Reprocessing Cycle--Growth Case 3, 1990 Reprocessing Startup, Watts(A)

RADIONUCLIDES (A)	Actinides											
	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)								
	2000	2040	2070	500	1000	5000	10000	50000	100000	500000	1000000	
CN=245	4.48E+01	2.26E+04	2.25E+04	2.18E+04	2.09E+04	1.49E+04	9.42E+03	3.43E+02	5.17E+00	0.	0.	
CN=244	1.68E+05	2.70E+07	1.26E+07	2.22E+00	1.07E+08	0.	0.	0.	0.	0.	0.	
CN=243	7.76E+02	3.72E+04	2.41E+04	3.65E+00	7.22E+05	0.	0.	0.	0.	0.	0.	
CN=242	2.59E+03	4.56E+05	4.16E+05	6.53E+04	6.88E+03	8.00E+05	0.	0.	0.	0.	0.	
AM=243+MP=239	4.42E+03	5.63E+05	5.44E+05	5.44E+05	5.20E+05	3.62E+05	2.20E+05	6.13E+03	6.60E+01	0.	0.	
AM=242+AM=247	1.38E+02	2.44E+04	2.23E+04	3.50E+03	3.48E+02	4.27E+06	0.	0.	0.	0.	0.	
AM=241	6.18E+05	1.35E+07	1.39E+07	7.92E+06	3.79E+06	2.22E+04	1.05E+04	3.45E+02	5.31E+00	0.	0.	
PU=242	9.13E+00	3.43E+03	3.43E+03	3.45E+03	3.45E+03	3.83E+03	3.79E+03	3.42E+03	3.22E+03	1.55E+03	6.20E+02	
PU=241	4.78E+02	4.90E+04	1.92E+04	2.88E+01	2.77E+01	1.98E+01	1.20E+01	4.44E-01	6.85E-03	0.	0.	
PU=240	3.03E+03	5.00E+05	5.14E+05	5.40E+05	5.14E+05	3.41E+05	2.04E+05	3.74E+03	2.00E+01	0.	0.	
PU=239	1.92E+04	1.31E+05	1.12E+05	1.15E+05	1.20E+05	1.45E+05	1.48E+05	7.40E+04	1.49E+04	2.19E+01	1.49E+07	
PU=238	1.12E+04	3.73E+06	3.35E+06	2.33E+05	1.44E+04	1.73E+04	0.	0.	0.	0.	0.	
PU=236	2.12E+01	2.24E+03	1.74E+01	0.	0.	0.	0.	0.	0.	0.	0.	
NP=237+PA=233	6.82E+01	4.19E+03	4.27E+03	5.43E+03	6.29E+03	6.96E+03	6.07E+03	6.02E+03	6.41E+03	5.98E+03	5.09E+03	
U=238+TH=234+ PA=234	1.50E+00	1.92E+01	1.92E+01	1.92E+01	1.92E+01	1.92E+01	1.92E+01	1.92E+01	1.93E+01	1.94E+01	1.95E+01	
U=236	6.01E+01	1.75E+01	1.76E+01	2.28E+01	2.08E+01	7.21E+01	1.06E+02	1.45E+02	1.56E+02	1.54E+02	1.52E+02	
U=235+TH=231	7.27E+02	4.44E+01	4.51E+01	9.48E+01	1.04E+00	1.52E+00	2.21E+00	4.40E+00	8.46E+00	9.05E+00	9.05E+00	
U=234	3.90E+00	3.14E+02	4.44E+02	1.48E+03	1.48E+03	1.68E+03	1.46E+03	1.40E+03	1.29E+03	4.32E+02	1.20E+02	
U=233	2.45E+03	5.49E+01	9.41E+01	4.33E+00	2.08E+01	1.30E+02	2.48E+02	1.36E+03	2.27E+03	5.28E+03	5.11E+03	
U=232	7.25E+02	1.52E+04	1.26E+04	2.52E+02	2.45E+00	0.	0.	0.	0.	0.	0.	
PA=231	6.01E+03	2.42E+01	2.02E+01	2.48E+01	2.27E+01	3.47E+01	4.15E+01	3.24E+00	6.46E+00	9.68E+00	9.68E+00	
TH=230	3.67E+02	5.24E+01	5.02E+01	4.23E+00	1.17E+01	6.72E+01	1.34E+02	5.04E+02	4.28E+02	5.86E+02	1.65E+02	
TH=229+7 DAUGHTERS	7.19E+06	7.24E+03	1.72E+02	1.09E+00	5.26E+00	1.74E+02	4.42E+02	6.08E+03	1.03E+04	3.65E+04	3.54E+04	
TH=228+4 DAUGHTERS	2.73E+00	4.95E+04	8.15E+04	1.47E+03	1.46E+01	6.74E+05	1.00E+04	1.49E+03	3.71E+03	1.97E+02	3.95E+02	
AC=227+7 DAUGHTERS	1.07E+02	1.04E+00	1.30E+00	1.51E+00	1.44E+00	2.27E+00	3.47E+00	2.14E+01	4.35E+01	6.32E+01	6.32E+01	
TH=227+2 DAUGHTERS	1.19E+04	3.00E+04	4.26E+04	4.30E+07	1.08E+06	9.63E+06	2.72E+05	2.01E+04	5.30E+04	2.82E+03	5.65E+03	
RA=226+5 DAUGHTERS	1.44E+03	5.29E+02	7.04E+02	1.91E+00	1.08E+01	2.19E+02	4.45E+02	3.07E+03	4.70E+03	3.30E+03	9.29E+02	
PR=210+2 DAUGHTERS	7.21E+05	5.56E+03	9.44E+03	3.82E+01	2.42E+00	4.71E+01	1.26E+02	6.40E+02	1.01E+03	7.10E+02	2.00E+02	
TOTAL	6.12E+05	4.61E+07	3.13E+07	9.06E+06	4.40E+06	8.98E+05	4.26E+05	1.11E+05	4.36E+04	5.46E+04	4.79E+04	

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.
 B. TH=229, 7 DAUGHTERS ARE RA=225, AC=225, FR=221, AT=217, BI=213, PB=209 AND TL=209 IS 9% OF TH=229 AND PO=213 IS 9% OF TH=229.
 TH=228, 6 DAUGHTERS ARE RA=224, RN=220, PO=216, PB=212, BI=212 AND TL=208 IS 34% OF TH=228 AND PO=212 IS 64% OF TH=228.
 AC=227, 7 DAUGHTERS ARE TH=227, RA=223, RN=219, PO=215, PB=211, BI=211 AND TL=207.
 TH=232, 2 DAUGHTERS ARE RA=228 AND AC=228.
 RA=226, 5 DAUGHTERS ARE RN=222, PO=218, PB=214, BI=214 AND PR=214.
 PB=210, 2 DAUGHTERS ARE BI=210 AND PO=210.

NOTE. IN ACCOUNTING FOR THE ACTIVITY IN THIS MANNER, BRANCHING DECAY IN THE CASE OF TL=208 (34%) = PO=212 (64%), AND TL=209 (9%) = PO=211 (9%) WERE COUNTED AS A SINGLE DAUGHTER IN EACH CASE. MINOR BRANCHING (1% OR LESS) WAS IGNORED.

TABLE A.3.7a. Heat Generation Rates, Reprocessing Cycle, Growth Case 3, 2010 Reprocessing Startup, Watts(A)

Fission and Activation Products

MAJOR RADIOISOTOPES	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	2040	2070	500	1000	5000	10000	50000	100000	500000	1000000
H-3	0.	6.76E+01	2.19E+01	2.58E+09	0.	0.	0.	0.	0.	0.	0.
C-10	0.	6.28E+01	6.26E+01	5.96E+01	5.41E+01	3.46E+01	1.89E+01	1.40E+01	3.96E+00	0.	0.
MN-54	0.	1.00E+00	5.61E+08	0.	0.	0.	0.	0.	0.	0.	0.
FE-55	0.	2.85E+03	1.38E+01	0.	0.	0.	0.	0.	0.	0.	0.
CO-60	0.	9.18E+04	2.24E+04	0.	0.	0.	0.	0.	0.	0.	0.
NI-59	0.	3.42E+01	3.42E+01	3.31E+01	3.39E+01	3.18E+01	3.05E+01	2.15E+01	1.40E+01	4.38E+01	5.77E+03
NI-63	0.	1.30E+04	1.12E+04	5.27E+02	1.02E+01	0.	0.	0.	0.	0.	0.
SE-79	0.	9.07E+01	3.06E+01	3.05E+01	3.04E+01	2.91E+01	2.78E+01	1.80E+01	1.06E+01	1.49E+01	7.22E+04
KR-85	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
RR-87	0.	2.45E+03	2.55E+03	2.55E+03	2.45E+03	2.55E+03	2.45E+03	2.45E+03	2.55E+03	2.55E+03	2.55E+03
BR-88+Y-90	0.	4.06E+07	2.08E+07	1.12E+03	4.06E+03	0.	0.	0.	0.	0.	0.
ZR-93	0.	4.80E+01	4.80E+01	4.80E+01	4.80E+01	4.79E+01	4.78E+01	4.80E+01	4.58E+01	3.81E+01	3.02E+01
NR-93M	0.	1.23E+02	1.35E+02	1.42E+02	1.02E+02	1.01E+02	1.01E+02	1.28E+02	1.35E+02	1.12E+02	8.93E+01
TC-99	0.	5.17E+03	5.17E+03	5.16E+03	5.15E+03	5.09E+03	5.00E+03	4.29E+03	3.72E+03	4.90E+02	1.91E+02
RU-106+DM-106	0.	6.25E+03	6.02E+03	0.	0.	0.	0.	0.	0.	0.	0.
PO-107	0.	2.04E+00	2.04E+00	2.04E+00	2.04E+00	2.04E+00	2.04E+00	2.03E+00	2.02E+00	1.90E+00	1.85E+00
AG-110M	0.	1.63E+00	3.76E+09	0.	0.	0.	0.	0.	0.	0.	0.
CO-113M	0.	6.94E+02	2.58E+02	4.87E+07	0.	0.	0.	0.	0.	0.	0.
BR-124+Y-125M	0.	1.94E+04	9.34E+01	0.	0.	0.	0.	0.	0.	0.	0.
SN-126+SS-126	0.	3.82E+03	3.82E+03	3.81E+03	3.79E+03	3.69E+03	3.46E+03	2.70E+03	1.01E+03	1.20E+02	3.75E+00
I-129	0.	5.21E+00	5.21E+00	5.21E+00	5.21E+00	5.21E+00	5.21E+00	5.20E+00	5.19E+00	5.11E+00	5.00E+00
CS-130	0.	2.06E+04	2.19E+02	0.	0.	0.	0.	0.	0.	0.	0.
CB-135	0.	3.26E+01	3.26E+01	3.26E+01	3.26E+01	3.25E+01	3.25E+01	3.22E+01	3.18E+01	2.90E+01	2.59E+01
CB-137+BA-137	0.	4.67E+07	2.94E+07	2.50E+03	2.02E+02	0.	0.	0.	0.	0.	0.
CE-140+PR-140	0.	9.76E+02	1.78E+05	0.	0.	0.	0.	0.	0.	0.	0.
PM-147	0.	2.02E+04	1.01E+02	0.	0.	0.	0.	0.	0.	0.	0.
SM-151	0.	3.46E+04	2.95E+05	1.16E+04	2.17E+02	0.	0.	0.	0.	0.	0.
EU-152	0.	6.74E+03	2.76E+03	1.43E+07	0.	0.	0.	0.	0.	0.	0.
EU-154	0.	2.32E+06	9.75E+05	2.27E+02	0.	0.	0.	0.	0.	0.	0.
EU-155	0.	3.55E+02	1.68E+01	0.	0.	0.	0.	0.	0.	0.	0.
OTMFR	0.	1.25E+02	0.	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	0.	9.05E+07	5.55E+07	2.51E+04	9.43E+03	9.10E+03	8.87E+03	7.45E+03	5.47E+03	1.30E+03	3.47E+02

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.

TABLE A.3.7b. Heat Generation Rates--Reprocessing Cycle--Growth Case 3 - 2010 Reprocessing Startup, Watts(A)

Actinides

RADIONUCLIDES (B)	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
CM-245	0.	9.95E+03	5.94E+03	5.74E+03	5.50E+03	3.93E+04	2.49E+03	0.13E+01	1.36E+00	0.	0.
CM-246	0.	9.86E+04	4.09E+06	7.93E-01	3.83E-09	0.	0.	0.	0.	0.	0.
CM-243	0.	2.13E+04	1.78E+04	2.09E+00	4.14E-05	0.	0.	0.	0.	0.	0.
CM-242	0.	1.09E+05	1.36E+05	2.14E+04	2.19E+03	2.62E-05	0.	0.	0.	0.	0.
AM-243+MP-239	0.	2.12E+05	2.11E+05	2.04E+05	1.05E+05	1.36E+05	4.42E+04	2.10E+03	2.07E+01	0.	0.
AM-242M+AM-242	0.	8.00E+03	7.30E+03	1.15E+03	1.17E+02	1.00E-06	0.	0.	0.	0.	0.
AM-241	0.	2.18E+07	2.12E+07	1.11E+07	4.07E+06	1.26E+04	2.76E+03	0.42E+01	1.03E+00	0.	0.
PU-242	0.	1.92E+02	1.93E+02	1.00E+02	1.42E+02	1.61E+02	1.40E+02	1.08E+02	1.35E+02	6.51E+01	2.61E+01
PU-241	0.	2.75E+03	1.08E+03	7.59E+00	7.39E+00	5.21E+00	7.02E+00	1.20E+01	1.80E+03	0.	0.
PU-240	0.	5.51E+04	6.73E+04	7.07E+04	7.11E+04	4.72E+04	2.43E+04	0.47E+02	2.77E+00	0.	0.
PU-239	0.	2.30E+04	2.31E+04	2.06E+04	2.47E+04	3.80E+04	6.09E+04	2.49E+04	5.08E+03	6.95E+02	4.71E+04
PU-238	0.	2.11E+05	1.09E+05	3.89E+04	4.41E+03	5.68E-05	0.	0.	0.	0.	0.
PU-236	0.	1.26E+01	9.72E-04	0.	0.	0.	0.	0.	0.	0.	0.
MP-237+PA-233	0.	2.90E+03	3.03E+03	4.71E+03	5.09E+03	6.96E+04	4.07E+03	5.49E+03	6.78E+03	5.95E+03	5.06E+03
U-238+TH-234+Pa-234m	0.	1.88E+01	1.88E+01	1.88E+01	1.88E+01	1.88E+01	1.88E+01	1.88E+01	1.88E+01	1.88E+01	1.89E+01
U-236	0.	1.30E+01	1.30E+01	1.37E+01	1.07E+01	2.05E+01	2.42E+01	1.51E+01	3.21E+01	3.18E+01	3.13E+01
U-235+TH-231	0.	9.56E+01	9.56E+01	9.64E-01	9.76E-01	1.09E+00	1.28E+00	2.43E+00	3.21E+00	3.40E+00	3.40E+00
U-234	0.	7.35E+01	8.45E+01	1.73E+02	2.07E+02	2.09E+02	2.06E+02	1.06E+02	1.64E+02	6.53E+01	2.93E+01
U-233	0.	2.25E+01	4.66E-01	6.27E+00	1.79E+01	1.26E+02	2.44E+02	1.55E+03	2.26E+03	5.25E+03	5.09E+03
U-232	0.	1.14E+00	9.47E-01	1.90E-02	1.44E-04	0.	0.	0.	0.	0.	0.
PA-231	0.	2.51E+01	2.51E+01	2.57E-01	2.46E-01	3.34E+01	4.29E+01	1.09E+00	2.64E+00	3.64E+00	3.64E+00
TH-230	0.	1.49E+00	1.50E+00	1.92E+00	2.41E+00	9.67E+00	1.40E+01	4.49E+01	1.05E+02	8.26E+01	3.43E+01
TH-229+7 DAUGHTERS	0.	1.64E+03	6.14E+03	7.46E-01	4.71E+00	1.65E+02	4.28E+02	4.00E+03	1.42E+04	3.63E+04	3.53E+04
TH-228+6 DAUGHTERS	0.	1.36E+01	6.27E+00	1.26E-01	1.03E-03	2.76E-05	6.19E-05	1.00E-04	4.17E-04	4.20E-03	8.38E-03
AC-227+7 DAUGHTERS	0.	7.49E+01	1.18E+00	1.67E+00	1.74E+00	2.18E+00	2.40E+00	0.71E+00	1.73E+01	2.38E+01	2.38E+01
TH-232+2 DAUGHTERS	0.	1.59E+07	4.00E+08	2.80E-07	6.47E-07	3.95E-06	4.40E-06	9.48E-05	1.17E+04	6.01E+04	1.20E+03
RA-226+5 DAUGHTERS	0.	1.25E+01	1.96E-01	1.54E+00	3.66E+00	3.38E+01	4.00E+01	1.49E+02	5.05E+02	4.65E+02	1.93E+02
PB-210+2 DAUGHTERS	0.	1.15E+02	2.30E-02	3.16E-01	8.41E-01	7.26E+00	1.72E+01	4.48E+01	1.08E+02	1.00E+02	4.15E+01
TOTAL	0.	3.21E+07	2.63E+07	1.14E+07	5.28E+06	2.45E+05	1.73E+05	4.29E+04	3.05E+04	4.84E+04	4.58E+04

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.

B. TH-229, 7 DAUGHTERS ARE RA-225, AC-225, PR-221, AT-217, RI-213, PB-209 AND TL-209 VS 9% OF TH-229 AND PO-213 IS 9% OF TH-229.

TH-228, 6 DAUGHTERS ARE RA-224, RN-220, PO-216, PB-212, RI-212 AND TL-208 IS 3% OF TH-228 AND PO-212 VS 6% OF TH-228.

AC-227, 7 DAUGHTERS ARE TH-227, RA-223, RN-219, PO-215, PA-211, RI-211 AND TL-207.

TH-232, 2 DAUGHTERS ARE RA-226 AND AC-226.

RA-226, 5 DAUGHTERS ARE RN-222, PN-218, PB-214, BI-214 AND Pn-214.

PB-210, 2 DAUGHTERS ARE BI-210 AND PO-210.

NOTE. IN ACCOUNTING FOR THE ACTIVITY IN THIS MANNER, BRANCHING DECAY IN THE CASE OF Tl-209 (34%) = PO-212 (64%), AND TL-209 (9%) = PO-211 (9%) WERE COUNTED AS A SINGLE DAUGHTER IN EACH CASE. MINOR BRANCHING (1% OR LESS) WAS IGNORED.

TABLE A.3.8a. Heat Generation Rates--Reprocessing Cycle--Growth Case 4 -- 2000 Reprocessing Startup, Watts(A)

Fission and Activation Products

MAJOR RADIONUCLIDES	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
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H-3	2.14E+01	1.33E+02	4.22E+01	5.06E+09	0.	0.	0.	0.	0.	0.	0.
C-14	1.01E+01	7.73E+01	7.72E+01	7.35E+01	6.02E+01	4.26E+01	2.43E+01	1.45E+01	4.49E+00	0.	0.
MN-54	2.89E+07	4.28E+00	4.63E+07	0.	0.	0.	0.	0.	0.	0.	0.
FE-55	3.62E+00	1.24E+04	6.25E+01	0.	0.	0.	0.	0.	0.	0.	0.
CO-60	9.55E+02	9.19E+05	6.74E+04	0.	0.	0.	0.	0.	0.	0.	0.
NI-59	6.68E+02	4.10E+01	4.10E+01	4.09E+01	4.27E+01	3.93E+01	3.76E+01	2.46E+01	1.73E+01	5.41E+01	7.13E+03
NI-63	2.93E+01	1.68E+04	1.04E+04	6.79E+02	1.47E+01	0.	0.	0.	0.	0.	0.
SE-79	3.25E+05	3.95E+01	3.05E+01	3.93E+01	3.01E+01	3.75E+01	3.46E+01	2.12E+01	1.36E+01	1.92E+01	9.31E+04
KR-85	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
RB-87	2.75E+09	3.19E+03	3.14E+03	3.19E+03	3.19E+03	3.19E+03	3.19E+03	3.19E+03	3.19E+03	3.19E+03	3.19E+03
BR-90+Y-90	5.98E+01	5.88E+07	3.54E+07	1.62E+03	7.18E+03	0.	0.	0.	0.	0.	0.
ZR-93	5.12E+03	6.14E+01	6.14E+01	6.14E+01	6.14E+01	6.13E+01	6.11E+01	6.00E+01	5.87E+01	4.84E+01	3.87E+01
NB-93M	1.24E+02	1.08E+02	1.44E+02	1.81E+02	1.81E+02	1.81E+02	1.81E+02	1.77E+02	1.73E+02	1.44E+02	1.14E+02
TC-99	9.98E+03	6.79E+03	6.79E+03	6.78E+03	6.77E+03	6.68E+03	6.57E+03	6.26E+03	4.84E+03	1.31E+03	2.51E+02
RU-106+PM-106	1.69E+05	5.59E+04	5.75E+02	0.	0.	0.	0.	0.	0.	0.	0.
PO-107	2.00E+06	2.96E+00	2.96E+00	2.96E+00	2.96E+00	2.96E+00	2.96E+00	2.95E+00	2.93E+00	2.82E+00	2.68E+00
AG-110M	0.	2.15E+01	4.04E+08	0.	0.	0.	0.	0.	0.	0.	0.
CO-113M	3.65E+01	1.09E+03	5.52E+02	1.04E+06	0.	0.	0.	0.	0.	0.	0.
BB-125+TE-125M	4.20E+02	7.94E+04	4.71E+02	0.	0.	0.	0.	0.	0.	0.	0.
SN-126+SB-126	3.80E+03	5.34E+03	5.34E+03	5.37E+03	5.35E+03	5.20E+03	5.03E+03	3.81E+03	2.70E+03	1.69E+02	5.29E+00
I-129	9.47E+03	7.05E+00	7.05E+00	7.04E+00	7.04E+00	7.04E+00	7.04E+00	7.03E+00	7.02E+00	6.90E+00	6.76E+00
CS-130	3.02E+02	1.26E+06	1.44E+03	0.	0.	0.	0.	0.	0.	0.	0.
CS-135	2.70E+05	4.74E+01	4.74E+01	4.74E+01	4.74E+01	4.73E+01	4.73E+01	4.44E+01	4.63E+01	4.22E+01	3.76E+01
CS-137+BA-137	6.47E+01	7.10E+07	4.47E+07	3.80E+03	3.48E+02	0.	0.	0.	0.	0.	0.
CE-144+PR-144	3.26E+08	7.75E+03	1.02E+04	0.	0.	0.	0.	0.	0.	0.	0.
PM-147	9.89E+03	4.72E+04	4.40E+02	0.	0.	0.	0.	0.	0.	0.	0.
SM-151	4.54E+01	4.93E+05	4.20E+05	1.66E+04	3.10E+02	0.	0.	0.	0.	0.	0.
EU-152	1.00E+02	2.06E+04	6.47E+03	4.31E+07	0.	0.	0.	0.	0.	0.	0.
EU-154	2.82E+00	4.41E+06	1.49E+06	4.31E+02	0.	0.	0.	0.	0.	0.	0.
EU-155	2.01E+05	2.38E+03	1.13E+00	0.	0.	0.	0.	0.	0.	0.	0.
OTHER	0.	1.06E+01	0.	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	1.12E+03	1.37E+08	6.71E+07	3.53E+04	1.29E+04	1.23E+04	1.20E+04	9.01E+03	7.90E+03	1.72E+03	4.56E+02

A. VALUES LESS THAN 1.0E+10 HAVE BEEN DESIGNATED AS ZERO.

TABLE A.3.8b. Heat Generation Rates--Reprocessing Cycle--Growth Case 4 - 2000 Reprocessing Startup, Watts(A)

Actinides

RADIONUCLIDES (A)	YEAR			GENLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
CM=245	4.02E+04	1.49E+04	1.49E+04	1.43E+04	1.75E+04	1.25E+04	8.24E+03	2.48E+02	4.74E+00	0.	0.
CM=244	2.36E+00	3.23E+07	1.50E+07	2.65E+00	1.28E+08	0.	0.	0.	0.	0.	0.
CM=243	1.26E+02	4.40E+04	2.99E+04	4.52E+00	8.04E+05	0.	0.	0.	0.	0.	0.
CM=242	2.79E+02	4.07E+05	3.72E+05	5.43E+04	5.06E+03	7.14E+04	0.	0.	0.	0.	0.
AM=243+AM=239	4.86E+02	4.98E+05	4.97E+05	4.79E+05	4.48E+05	3.19E+05	2.03E+05	5.40E+03	5.82E+01	0.	0.
AM=242+AM=240	2.77E+03	2.18E+04	1.99E+04	3.12E+03	3.19E+02	3.81E+06	0.	0.	0.	0.	0.
AM=241	2.43E+01	2.03E+07	1.08E+07	1.04E+07	4.48E+06	2.17E+04	4.79E+03	1.06E+02	4.42E+00	0.	0.
PU=242	9.06E+02	3.33E+02	3.35E+02	3.55E+02	3.40E+02	3.54E+02	3.85E+02	3.30E+02	3.01E+02	1.44E+02	5.40E+01
PU=241	4.02E+00	7.31E+03	2.88E+03	2.42E+01	2.72E+01	1.66E+01	1.49E+01	3.41E+01	5.75E+03	0.	0.
PU=240	3.79E+01	1.49E+05	1.49E+05	2.16E+05	2.45E+05	1.36E+05	4.16E+04	1.35E+03	4.01E+00	0.	0.
PU=239	2.80E+01	3.74E+04	3.76E+04	4.14E+04	4.49E+04	7.50E+04	4.40E+04	5.22E+04	1.31E+04	1.52E+01	1.03E+07
PU=238	1.04E+02	6.81E+05	6.33E+05	1.10E+05	1.26E+04	1.55E+04	0.	0.	0.	0.	0.
PU=236	3.41E+03	9.49E+01	7.32E+03	0.	0.	0.	0.	0.	0.	0.	0.
NP=237+PA=233	2.19E+03	5.25E+03	5.37E+03	6.97E+03	8.16E+03	9.07E+03	9.84E+03	9.00E+03	4.45E+03	7.74E+03	6.61E+03
U=238+TH=234+ Pa=234M	2.17E+02	2.43E+01	2.53E+01	2.53E+01	2.43E+01	2.53E+01	2.43E+01	2.43E+01	2.43E+01	2.53E+01	2.53E+01
U=236	4.61E+03	2.20E+01	2.21E+01	2.41E+01	2.49E+01	4.34E+01	5.74E+01	7.71E+01	7.74E+01	7.65E+01	7.54E+01
U=235+TH=231	1.10E+03	1.25E+00	1.25E+00	1.27E+00	1.29E+00	1.51E+00	1.89E+00	4.78E+00	4.06E+00	6.48E+00	6.47E+00
U=234	4.37E+02	1.26E+02	1.58E+02	4.28E+02	5.23E+02	5.28E+02	4.21E+02	4.49E+02	4.10E+02	1.49E+02	5.46E+01
U=233	4.08E+06	4.44E+01	9.14E+01	1.02E+01	2.44E+01	1.68E+02	3.08E+02	1.44E+03	2.95E+03	6.86E+03	6.63E+03
U=232	4.87E+04	2.63E+00	2.20E+00	4.42E+02	3.48E+04	0.	0.	0.	0.	0.	0.
PA=231	2.43E+07	3.32E+01	3.32E+01	3.40E+01	3.41E+01	4.43E+01	5.43E+01	2.41E+00	4.44E+00	6.93E+00	6.92E+00
TH=230	1.89E+06	1.08E+00	1.10E+00	2.11E+00	4.77E+00	2.18E+01	4.29E+01	1.72E+02	2.42E+02	1.96E+02	6.80E+01
TH=229+7 DAUGHTERS	2.67E+10	4.23E+03	1.34E+02	1.27E+00	7.28E+00	2.24E+02	4.34E+02	9.04E+03	1.46E+04	4.75E+04	4.61E+04
TH=228+6 DAUGHTERS	5.72E+04	1.98E+01	1.06E+01	2.92E+01	2.48E+03	4.74E+05	1.07E+04	4.20E+04	1.02E+03	7.35E+03	1.47E+02
AC=227+7 DAUGHTERS	9.22E+09	1.16E+00	1.64E+00	2.21E+00	2.40E+00	2.90E+00	4.41E+00	1.44E+01	3.17E+01	4.53E+01	4.52E+01
TH=232+2 DAUGHTERS	0.	4.70E+08	4.52E+08	4.83E+07	1.14E+06	6.77E+06	1.42E+05	9.71E+05	2.04E+04	1.05E+03	2.09E+03
RA=226+3 DAUGHTERS	4.63E+08	9.77E+02	1.50E+01	1.40E+00	4.47E+00	7.23E+03	1.48E+02	9.73E+02	1.49E+03	1.10E+03	3.43E+02
PB=210+2 DAUGHTERS	3.66E+09	9.50E+03	1.40E+02	2.87E+01	1.05E+00	1.55E+01	4.04E+01	2.09E+02	3.20E+02	2.37E+02	8.23E+01
TOTAL	2.01E+02	5.49E+07	3.66E+07	1.13E+07	5.03E+06	5.75E+05	4.06E+05	4.15E+04	4.65E+04	6.41E+04	6.01E+04

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.
 B. TH=229, 7 DAUGHTERS ARE RA=225, AC=223, FR=221, AT=217, RI=213, PB=209 AND TL=204 IS 9% OF TH=229 AND PD=213 IS 91% OF TH=229.
 TH=228, 6 DAUGHTERS ARE RA=224, RN=220, PD=216, PB=212, RI=212 AND TL=204 IS 34% OF TH=228 AND PD=213 IS 60% OF TH=228.
 AC=227, 7 DAUGHTERS ARE TH=227, RI=223, RN=219, PD=215, RA=211, RI=211 AND TL=207.
 TH=232, 2 DAUGHTERS ARE RA=228 AND AC=228.
 PA=226, 3 DAUGHTERS ARE RN=222, RN=214, PB=214, BI=214 AND PD=214.
 PB=210, 2 DAUGHTERS ARE RI=210 AND PD=210.

NOTE. IN ACCOUNTING FOR THE ACTIVITY IN THIS MANNER, BRANCHING DECAY IN THE CASE OF TI=208 (34%) - PD=212 (64%), AND TL=209 (9%) - PD=211 (91%) WERE COUNTED AS A SINGLE DAUGHTER IN EACH CASE. MINOR BRANCHING (1% OR LESS) WAS IGNORED.

TABLE A.3.9a. Heat Generation Rates--Reprocessing Cycle--Growth Case 5 - 2000 Reprocessing Startup, Watts(A)

Fission and Activation Products

HAJRA RADIOISOTOPES	YEAR			REACTOR TYPE (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000

H-3	2.14E+01	2.16E+02	6.94E+01	8.21E+09	0.	0.	0.	0.	0.	0.	0.
C-14	1.41E+01	1.05E+02	1.04E+02	9.43E+01	9.75E+01	5.76E+01	7.15E+01	2.40E+01	5.93E+00	0.	0.
MN-54	2.89E+07	1.56E+01	8.74E+07	0.	0.	0.	0.	0.	0.	0.	0.
FE-55	3.62E+00	2.37E+04	1.15E+02	0.	0.	0.	0.	0.	0.	0.	0.
CO-60	9.55E+02	1.67E+04	1.27E+05	0.	0.	0.	0.	0.	0.	0.	0.
NI-63	4.68E+02	5.53E+01	5.53E+01	5.51E+01	5.49E+01	5.30E+01	4.04E+01	1.20E+01	2.73E+01	7.29E+01	9.41E+03
NI-64	2.93E+01	2.33E+04	2.00E+04	9.42E+02	2.14E+01	0.	0.	0.	0.	0.	0.
SE-79	7.25E+05	5.45E+01	5.45E+01	5.42E+01	5.40E+01	5.27E+01	4.01E+01	1.14E+01	1.84E+01	2.60E+01	1.26E+03
KR-85	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
RB-87	2.75E+09	4.30E+03	4.30E+03	4.30E+03	4.30E+03	4.30E+03	4.30E+03	4.30E+03	4.30E+03	4.30E+03	4.30E+03
SR-90+Y-90	4.98E+01	4.62E+07	5.07E+07	2.37E+03	1.05E+02	0.	0.	0.	0.	0.	0.
ZR-93	5.12E+03	4.30E+01	4.30E+01	4.30E+01	4.30E+01	4.24E+01	4.24E+01	4.11E+01	7.03E+01	6.59E+01	5.23E+01
NB-93M	1.24E+02	1.91E+02	2.26E+02	2.45E+02	2.49E+02	2.45E+02	2.44E+02	2.10E+02	2.74E+02	1.95E+02	1.84E+02
TC-99	9.08E+03	9.22E+03	9.22E+03	9.21E+03	9.20E+03	9.08E+03	8.03E+03	7.12E+03	6.43E+03	1.77E+03	3.41E+02
RU-106+RH-106	1.65E+05	1.07E+09	1.10E+01	0.	0.	0.	0.	0.	0.	0.	0.
PD-107	2.00E+06	4.10E+00	4.10E+00	4.10E+00	4.10E+00	4.10E+00	4.10E+00	4.04E+00	4.06E+00	3.90E+00	3.72E+00
AG-110M	0.	4.20E+01	4.46E+08	0.	0.	0.	0.	0.	0.	0.	0.
CO-113M	3.65E+01	2.39E+03	8.47E+02	1.47E+06	0.	0.	0.	0.	0.	0.	0.
BB-125+TE-125M	4.20E+02	1.50E+05	8.44E+02	0.	0.	0.	0.	0.	0.	0.	0.
BN-126+BR-126	3.80E+03	7.42E+03	7.41E+03	7.39E+03	7.37E+03	7.17E+03	6.02E+03	5.25E+03	3.71E+03	2.33E+02	7.29E+00
I-129	9.97E+03	9.42E+00	9.42E+00	9.42E+00	9.42E+00	9.42E+00	9.42E+00	9.42E+00	9.42E+00	9.42E+00	9.24E+00
CS-134	3.02E+02	2.37E+06	2.75E+03	0.	0.	0.	0.	0.	0.	0.	0.
CR-135	2.70E+05	4.82E+01	6.42E+01	6.51E+01	6.51E+01	6.31E+01	4.40E+01	4.44E+01	4.37E+01	5.81E+01	5.17E+01
CS-137+CA-137	4.47E+01	1.05E+04	6.44E+07	5.59E+03	5.02E+02	0.	0.	0.	0.	0.	0.
CE-144+PR-144	3.26E+08	1.44E+04	2.63E+04	0.	0.	0.	0.	0.	0.	0.	0.
PM-147	9.89E+03	1.57E+05	7.45E+02	0.	0.	0.	0.	0.	0.	0.	0.
SM-151	4.54E+01	6.92E+04	5.97E+05	2.33E+04	4.25E+02	0.	0.	0.	0.	0.	0.
EU-152	1.00E+02	3.35E+04	1.76E+04	7.00E+07	0.	0.	0.	0.	0.	0.	0.
EU-154	2.82E+00	4.94E+04	2.92E+06	5.78E+02	0.	0.	0.	0.	0.	0.	0.
EU-155	2.41E+05	4.53E+03	2.15E+00	0.	0.	0.	0.	0.	0.	0.	0.
OTHER	0.	2.00E+01	1.04E+10	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	1.12E+03	2.03E+09	1.22E+08	4.94E+04	1.76E+04	1.48E+04	1.44E+04	1.35E+04	1.08E+04	2.34E+03	6.20E+02

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.

TABLE A.3.9b. Heat Generation Rates--Reprocessing Cycle--Growth Case 5 - 2000 Reprocessing Startup, Watts(A)

Actinides

RADIONUCLIDES (R)	YEAR			GENERATION TIME (YEARS) BEYOND 1975							
	2000	2050	2170	500	1000	5000	10000	50000	100000	500000	1000000
CM-245	4.02E+04	3.00E+04	3.20E+04	2.97E+04	2.78E+04	1.99E+04	1.81E+04	4.56E+02	4.84E+00	0.	0.
CM-244	2.36E+00	5.55E+07	2.34E+07	4.58E+00	2.20E+05	0.	0.	0.	0.	0.	0.
CM-243	1.26E+02	7.20E+04	4.57E+04	7.07E+00	1.00E+04	0.	0.	0.	0.	0.	0.
CM-242	2.79E+02	6.44E+05	5.92E+05	9.28E+04	9.09E+13	1.14E+04	0.	0.	0.	0.	0.
AM-243+MP-239	4.86E+02	7.72E+05	7.71E+05	7.43E+05	7.10E+05	4.94E+05	3.14E+05	4.27E+03	2.02E+01	0.	0.
AM-242+AM-242	2.77E+03	3.47E+04	3.17E+04	4.97E+03	5.08E+02	6.07E+04	0.	0.	0.	0.	0.
AM-241	2.43E+01	2.79E+07	2.74E+07	1.44E+07	6.07E+06	3.28E+04	1.49E+04	4.96E+02	7.33E+00	0.	0.
PU-242	9.06E+02	6.77E+02	6.01E+02	7.12E+02	7.20E+02	7.16E+02	7.10E+02	6.40E+02	6.02E+02	2.90E+02	1.16E+02
PU-241	4.02E+00	1.63E+04	6.07E+03	3.83E+01	3.69E+01	2.63E+01	1.73E+01	6.04E+01	9.11E+03	0.	0.
PU-240	3.79E+01	2.29E+05	2.94E+05	3.45E+05	3.29E+05	2.18E+05	1.81E+05	2.76E+03	1.28E+01	0.	0.
PU-239	2.80E+01	5.60E+04	5.63E+04	6.21E+04	7.01E+04	1.34E+05	1.03E+05	4.05E+04	2.02E+04	2.35E+01	1.59E+07
PU-238	1.04E+02	1.07E+04	9.92E+03	1.74E+03	2.01E+04	2.46E+04	0.	0.	0.	0.	0.
PU-236	3.41E+03	2.82E+03	2.14E+01	0.	0.	0.	0.	0.	0.	0.	0.
NP-237+PA-233	2.19E+03	6.49E+03	6.65E+03	8.84E+03	1.05E+04	1.18E+04	1.19E+04	1.17E+04	1.15E+04	1.01E+04	8.59E+03
U-238+TH-234+P-234M	2.17E+02	3.44E+01	3.04E+01	3.04E+01	3.04E+01	3.44E+01	3.04E+01	3.04E+01	3.04E+01	3.45E+01	3.05E+01
U-236	4.61E+03	2.70E+01	2.75E+01	3.07E+01	3.02E+01	6.22E+01	4.00E+01	1.76E+02	1.16E+02	1.15E+02	1.13E+02
U-235+TH-231	1.10E+03	1.65E+00	1.45E+00	1.67E+00	1.70E+00	2.03E+00	2.42E+00	7.07E+00	9.05E+00	9.64E+00	9.68E+00
U-234	4.37E+02	1.60E+02	2.19E+02	6.35E+02	7.07E+02	7.96E+02	7.05E+02	7.05E+02	6.17E+02	2.22E+02	7.00E+01
U-233	4.08E+04	5.58E+01	1.09E+00	1.27E+01	3.45E+01	2.17E+02	4.81E+02	2.73E+03	3.44E+03	8.92E+03	8.44E+03
U-232	4.87E+04	6.94E+02	6.61E+02	1.33E+01	1.08E+01	0.	0.	0.	0.	0.	0.
PA-231	2.43E+07	4.14E+01	4.14E+01	4.25E+01	4.00E+01	5.65E+01	7.60E+01	3.63E+00	7.17E+00	1.04E+01	1.03E+01
TH-230	1.89E+06	1.32E+00	1.35E+00	2.41E+00	6.20E+00	3.25E+01	6.02E+01	2.59E+02	3.99E+02	2.93E+02	9.93E+01
TH-229+7 DAUGHTERS	2.67E+10	4.61E+03	1.54E+02	1.57E+00	9.17E+00	2.88E+02	1.08E+03	1.78E+04	2.01E+04	6.17E+04	5.99E+04
TH-228+6 DAUGHTERS	5.72E+04	4.17E+03	4.76E+03	8.79E+01	7.13E+01	6.16E+05	1.02E+04	9.09E+04	2.01E+03	1.00E+02	2.07E+02
AC-227+7 DAUGHTERS	5.22E+09	1.37E+00	2.00E+00	2.76E+00	2.88E+00	3.49E+00	4.07E+00	2.47E+01	4.69E+01	6.77E+01	6.76E+01
TH-232+2 DAUGHTERS	0.	7.83E+08	5.40E+04	5.99E+07	1.03E+06	8.80E+04	2.03E+05	1.76E+04	2.87E+04	1.09E+03	2.97E+03
RA-226+5 DAUGHTERS	6.63E+08	1.12E+01	1.75E+01	1.78E+00	6.44E+00	1.07E+02	2.01E+02	1.04E+03	2.24E+03	1.65E+03	5.59E+02
PB-210+2 DAUGHTERS	3.66E+09	1.05E+02	2.07E+02	3.62E+01	1.03E+00	2.30E+01	6.04E+01	3.74E+02	4.81E+02	3.50E+02	1.20E+02
TOTAL	2.01E+02	8.64E+07	5.80E+07	1.58E+07	7.65E+06	8.94E+04	4.70E+05	1.03E+05	6.43E+04	9.37E+04	7.83E+04

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.
B. TH-229, 7 DAUGHTERS ARE RA-225, AC-225, FR-221, AT-217, RI-213, PB-209 AND TL-200 VS 9% OF TH-229 AND PD-213 IS 91% OF TH-229.
TH-228, 6 DAUGHTERS ARE RA-224, RN-220, PU-216, PB-212, PI-212 AND TL-208 IS 34% OF TH-228 AND PD-213 IS 66% OF TH-228.
AC-227, 7 DAUGHTERS ARE TH-227, RA-223, RN-219, PD-215, PA-211, AT-211 AND TL-207.
TH-232, 2 DAUGHTERS ARE RA-223 AND AC-228.
PA-226, 5 DAUGHTERS ARE RN-222, PD-213, PD-214, BI-214 AND PD-214.
PB-210, 2 DAUGHTERS ARE RI-210 AND PD-210.

NOTE. IN ACCOUNTING FOR THE ACTIVITY IN THIS MANNER, BRANCHING DECAY IN THE CASE OF TI-208 (34%) = PD-212 (64%), AND TL-209
----- (9%) = PD-211 (91%) WERE COUNTED AS A SINGLE DAUGHTER IN EACH CASE. MINOR BRANCHING (1% OR LESS) WAS IGNORED.

A.4' HAZARD INDEX TABLES

The tables of hazard indices (A.4.1a through A.4.9b) appear in the same format as those for the radioactivity inventory (A.2) and heat generation rates (A.3).

The hazard index employed here is the amount of water (m^3) required to dilute the quantity of a radionuclide present in one metric ton of spent fuel (MTHM) to drinking water standards. Following the summation at the bottom of each table, a uranium ore index is also shown. This is the ratio of the hazard index for the spent fuel to the hazard index ($8.7 \times 10^7 m^3$) for the quantity of 0.2% U_3O_8 uranium ore required to produce one metric ton of 3% ^{235}U fuel (see Section 3.4 for further discussion of these indices).

The total index for the fission products and activation products must be added to the total index for the actinides to obtain the total spent fuel index.

TABLE A.4.1a. Hazard Index--Once-Through Cycle--Growth Case 1, m³ water/MTHM(A)

Fission and Activation Products

MAJOR RADIONUCLIDES	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
H-3	0.	0.	6.55E+02	7.70E+08	0.	0.	0.	0.	0.	0.	0.
C-14	0.	0.	7.64E+02	7.28E+02	6.25E+02	4.22E+02	2.31E+02	1.23E+00	4.55E+03	0.	0.
MN-54	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
FE-55	0.	0.	9.13E+04	0.	0.	0.	0.	0.	0.	0.	0.
CO-60	0.	0.	7.67E+02	0.	0.	0.	0.	0.	0.	0.	0.
NI-63	0.	0.	1.16E+04	1.15E+04	1.15E+04	1.11E+04	1.06E+04	7.42E+03	4.08E+03	1.53E+02	2.01E+00
NI-63	0.	0.	6.25E+06	2.94E+05	6.21E+03	5.67E+10	0.	0.	0.	0.	0.
SE-79	0.	0.	8.74E+04	8.70E+04	8.45E+04	8.29E+04	7.46E+04	5.73E+04	3.01E+04	4.24E+02	2.06E+00
KR-85	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
RB-87	0.	0.	1.31E+01	1.31E+01	1.31E+01	1.31E+01	1.41E+01	1.41E+01	1.31E+01	1.31E+01	1.31E+01
SR-90+Y-90	0.	0.	1.87E+10	8.40E+05	3.73E+00	0.	0.	0.	0.	0.	0.
ZR-93	0.	0.	1.96E+03	1.56E+03	1.46E+03	1.56E+03	1.46E+03	1.43E+03	1.49E+03	1.24E+03	9.85E+02
NB-93M	0.	0.	3.10E+03	3.13E+03	3.13E+03	3.12E+03	4.12E+03	3.06E+03	2.99E+03	2.49E+03	1.97E+03
TC-99	0.	0.	3.22E+04	3.22E+04	3.22E+04	3.17E+04	4.12E+04	2.74E+04	2.32E+04	6.20E+03	1.10E+03
RU-106+RH-106	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
PD-107	0.	0.	2.45E+04	2.35E+04	2.45E+04	2.35E+04	2.45E+04	2.44E+04	2.33E+04	2.24E+04	2.13E+04
AG-110M	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
CD-113M	0.	0.	1.97E+05	3.72E+04	0.	0.	0.	0.	0.	0.	0.
BB-125+BE-125M	0.	0.	1.31E+02	0.	0.	0.	0.	0.	0.	0.	0.
BN-126+SB-126	0.	0.	3.44E+05	3.43E+05	3.42E+05	3.33E+05	3.21E+05	2.43E+05	1.72E+05	1.08E+04	3.38E+02
I-129	0.	0.	4.09E+05	4.09E+05	4.09E+05	4.09E+05	4.09E+05	4.09E+05	4.08E+05	4.01E+05	3.93E+05
CS-130	0.	0.	3.27E+01	0.	0.	0.	0.	0.	0.	0.	0.
CS-135	0.	0.	1.92E+03	1.92E+03	1.92E+03	1.92E+03	1.01E+03	1.40E+03	1.68E+03	1.71E+03	1.52E+03
CS-137+CA-137	0.	0.	4.26E+08	3.62E+04	3.41E+01	0.	0.	0.	0.	0.	0.
CE-140+ER-144	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
PM-147	0.	0.	8.16E+02	0.	0.	0.	0.	0.	0.	0.	0.
SM-151	0.	0.	1.15E+06	4.55E+04	8.09E+02	0.	0.	0.	0.	0.	0.
EU-152	0.	0.	7.07E+02	4.69E+08	0.	0.	0.	0.	0.	0.	0.
EU-154	0.	0.	3.62E+06	8.42E+02	0.	0.	0.	0.	0.	0.	0.
EU-155	0.	0.	1.42E+06	0.	0.	0.	0.	0.	0.	0.	0.
DMFR	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	0.	0.	1.91E+10	2.13E+06	9.20E+05	8.98E+05	4.41E+05	7.44E+05	6.68E+05	4.46E+05	4.20E+05
URANIUM ORE INDEX	0.	0.	2.19E+02	2.45E+02	1.06E+02	1.03E+02	1.01E+02	8.43E+03	7.68E+03	5.13E+03	4.83E+03

A. VALUES LESS THAN 1.0E+10 HAVE BEEN DESIGNATED AS ZERO.

TABLE A.4.1b. Hazard Index--Once-Through Cycle--Growth Case 1, m³ water/MTHM(A)

Actinides

RAOYDNUCLIDES (A)	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
CN=245	0.	0.	2.26E+04	2.18E+04	2.09E+04	1.50E+04	9.04E+03	7.83E+02	5.18E+00	0.	0.
CN=244	0.	0.	2.03E+06	5.18E+01	2.50E+09	0.	0.	0.	0.	0.	0.
CN=243	0.	0.	6.10E+04	9.24E+00	1.43E+04	0.	0.	0.	0.	0.	0.
CN=242	0.	0.	1.02E+05	2.85E+04	2.01E+03	3.49E+05	0.	0.	0.	0.	0.
AM=243+AM=239	0.	0.	2.02E+06	1.05E+06	1.86E+06	1.30E+06	8.24E+05	2.50E+04	2.76E+02	0.	0.
AM=242+AM=242	0.	0.	1.15E+06	1.81E+05	1.45E+04	2.21E+04	0.	0.	0.	0.	0.
AM=241	0.	0.	6.31E+08	3.35E+08	1.50E+08	2.65E+05	9.04E+03	7.83E+02	5.19E+00	0.	0.
PU=242	0.	0.	2.00E+05	1.09E+05	1.09E+05	1.98E+05	1.06E+05	1.22E+05	1.66E+05	9.00E+04	3.21E+04
PU=241	0.	0.	5.82E+06	4.37E+02	4.19E+02	3.00E+02	1.07E+02	6.88E+00	1.04E+01	0.	0.
PU=240	0.	0.	7.92E+07	7.02E+07	6.87E+07	4.03E+07	2.85E+07	4.29E+05	2.60E+03	0.	0.
PU=239	0.	0.	5.43E+07	5.37E+07	5.20E+07	4.74E+07	4.12E+07	1.33E+07	3.22E+06	3.74E+01	2.53E+05
PU=238	0.	0.	1.27E+08	5.57E+06	1.76E+05	3.37E+04	0.	0.	0.	0.	0.
PU=236	0.	0.	3.10E+05	0.	0.	0.	0.	0.	0.	0.	0.
NP=237+PA=233	0.	0.	9.00E+04	1.76E+05	2.20E+05	2.72E+05	2.72E+05	2.88E+05	2.64E+05	2.32E+05	1.97E+05
U=238+TH=234+ Pa=234	0.	0.	2.36E+04	2.38E+04	2.28E+04	2.38E+04	2.28E+04	2.38E+04	2.38E+04	2.38E+04	2.38E+04
U=236	0.	0.	5.90E+03	6.03E+03	6.20E+03	7.25E+03	8.09E+03	9.21E+03	9.32E+03	9.21E+03	9.08E+03
U=235+TH=231	0.	0.	8.25E+02	8.29E+02	8.24E+02	8.72E+02	9.13E+02	1.10E+03	1.17E+03	1.19E+03	1.19E+03
U=234	0.	0.	3.82E+04	4.52E+04	4.88E+04	4.54E+04	4.50E+04	4.13E+04	3.73E+04	1.93E+04	1.27E+04
U=233	0.	0.	4.05E+00	2.74E+01	7.10E+01	5.05E+02	1.05E+03	5.00E+03	8.09E+03	2.09E+04	2.03E+04
U=232	0.	0.	1.64E+03	3.30E+01	2.87E+01	0.	0.	0.	0.	0.	0.
PA=231	0.	0.	6.28E+01	2.80E+02	5.18E+02	2.49E+03	4.85E+03	1.02E+04	2.83E+04	3.43E+04	3.43E+04
TH=230	0.	0.	4.09E+02	2.60E+03	5.82E+03	2.47E+04	4.63E+04	2.59E+05	3.55E+05	3.39E+05	2.06E+05
TH=229+7 DAUGHTERS	0.	0.	7.89E+01	1.73E+01	8.54E+01	2.88E+03	1.09E+04	1.21E+05	2.47E+05	6.30E+05	6.12E+05
TH=228+6 DAUGHTERS	0.	0.	3.92E+03	7.07E+01	5.20E+01	3.37E+02	7.24E+02	4.24E+01	9.22E+01	4.72E+00	9.41E+00
AC=227+7 DAUGHTERS	0.	0.	8.22E+01	4.59E+02	9.25E+02	4.05E+03	8.88E+03	3.20E+04	5.07E+04	6.13E+04	6.13E+04
TH=232+2 DAUGHTERS	0.	0.	2.00E+02	1.39E+01	2.05E+01	1.65E+00	4.98E+00	2.16E+01	4.50E+01	2.30E+02	4.59E+02
RA=226+5 DAUGHTERS	0.	0.	5.15E+02	1.62E+04	8.26E+04	1.14E+04	2.05E+06	1.25E+07	2.41E+07	2.29E+07	1.39E+07
PB=210+2 DAUGHTERS	0.	0.	9.44E+01	5.39E+03	2.24E+04	3.89E+05	1.00E+06	5.26E+06	8.20E+06	7.78E+06	4.72E+06
TOTAL	0.	0.	8.98E+06	4.67E+08	2.73E+08	9.54E+07	7.32E+07	3.24E+07	3.67E+07	3.21E+07	1.98E+07
URANIUM ORE INDEX	0.	0.	1.03E+01	5.37E+00	3.13E+00	1.10E+00	8.01E+01	4.07E+01	4.22E+01	3.69E+01	2.26E+01

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.
 B. TH=229, 7 DAUGHTERS ARE RA=225, AC=225, PR=221, AT=217, AI=213, PB=209 AND TL=209 IS 9% OF TH=229 AND PO=213 IS 91% OF TH=229.
 TH=228, 6 DAUGHTERS ARE RA=224, RN=220, PO=216, PB=212, AI=212 AND TL=209 IS 36% OF TH=228 AND PB=212 IS 64% OF TH=228.
 AC=227, 7 DAUGHTERS ARE TH=227, RA=223, RN=219, PO=215, PB=211, BI=211 AND TL=207.
 TH=222, 2 DAUGHTERS ARE RA=228 AND AC=228.
 PA=226, 5 DAUGHTERS ARE RN=222, PO=214, PB=214, BI=214 AND PN=214.
 PB=210, 2 DAUGHTERS ARE BI=210 AND PO=210.

NOTE. IN ACCOUNTING FOR THE ACTIVITY IN THIS MANNER, BRANCHING DECAY IN THE CASE OF TL=209 (3%) = PO=212 (64%), AND TL=209
 (9%) = PO=213 (91%) WERE COUNTED AS A SINGLE DAUGHTER IN EACH CASE. MINOR BRANCHING (1% OR LESS) WAS IGNORED.

TABLE A.4.2a. Hazard Index--Once-Through Cycle--Growth Case 2, m³ water/MTHM(A)

Fission and Activation Products

MAJOR RADYONUCLIDES	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
H=3	0.	0.	2.34E+03	2.80E+07	0.	0.	0.	0.	0.	0.	0.
C=10	0.	0.	9.06E+02	9.48E+02	8.02E+02	5.50E+02	7.01E+02	2.79E+00	5.66E+03	0.	0.
MN=54	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
FE=55	0.	0.	4.29E-01	0.	0.	0.	0.	0.	0.	0.	0.
CO=60	0.	0.	1.40E+04	0.	0.	0.	0.	0.	0.	0.	0.
NI=69	0.	0.	1.09E+04	1.08E+04	1.07E+04	1.02E+04	1.76E+04	4.25E+03	6.26E+03	1.96E+02	2.58E+00
NI=63	0.	0.	8.99E+06	4.23E+05	9.20E+03	8.16E+10	0.	0.	0.	0.	0.
SE=79	0.	0.	1.14E+05	1.14E+05	1.13E+05	1.08E+05	1.73E+05	4.70E+04	3.94E+04	5.34E+02	2.69E+00
KR=85	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
RB=87	0.	0.	1.69E-01	1.69E-01	1.69E-01	1.69E-01	1.69E-01	1.69E-01	1.69E-01	1.69E-01	1.69E-01
BR=90+Y=90	0.	0.	3.60E+10	1.62E+06	7.19E+00	0.	0.	0.	0.	0.	0.
ZR=93	0.	0.	2.03E+03	2.03E+03	2.03E+03	2.03E+03	2.03E+03	1.49E+03	1.44E+03	1.61E+03	1.28E+03
NB=93M	0.	0.	3.08E+03	4.07E+03	4.07E+03	4.06E+03	4.05E+03	7.99E+03	3.49E+03	3.23E+03	2.57E+03
TC=99	0.	0.	4.20E+04	4.19E+04	4.18E+04	4.13E+04	4.06E+04	3.46E+04	3.02E+04	8.07E+03	1.55E+03
RU=106+PM=106	0.	0.	1.74E-07	0.	0.	0.	0.	0.	0.	0.	0.
PD=107	0.	0.	3.19E+04	3.19E+04	3.19E+04	3.19E+04	7.19E+04	3.17E+04	3.16E+04	3.04E+04	2.89E+04
AG=110M	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
CD=113M	0.	0.	7.76E+05	1.46E+03	0.	0.	0.	0.	0.	0.	0.
BR=125+Y=125M	0.	0.	6.10E+00	0.	0.	0.	0.	0.	0.	0.	0.
BN=126+BB=126	0.	0.	4.63E+05	4.62E+05	4.60E+05	4.48E+05	4.73E+05	7.59E+05	2.12E+05	1.49E+04	4.56E+02
I=129	0.	0.	5.00E+05	5.00E+05	5.00E+05	5.00E+05	5.39E+05	5.49E+05	5.37E+05	5.29E+05	5.18E+05
CS=134	0.	0.	1.51E+03	0.	0.	0.	0.	0.	0.	0.	0.
CS=135	0.	0.	2.63E+03	2.63E+03	2.63E+03	2.63E+03	2.62E+03	2.60E+03	2.57E+03	2.34E+03	2.09E+03
CS=137+BA=137	0.	0.	8.15E+08	6.92E+04	6.70E+01	0.	0.	0.	0.	0.	0.
CE=144+PR=144	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
PM=147	0.	0.	2.99E+01	0.	0.	0.	0.	0.	0.	0.	0.
BN=151	0.	0.	1.46E+06	6.17E+04	1.15E+03	1.70E+11	0.	0.	0.	0.	0.
EU=152	0.	0.	2.49E+03	1.72E+07	0.	0.	0.	0.	0.	0.	0.
EU=154	0.	0.	1.20E+07	2.80E+01	1.11E+10	0.	0.	0.	0.	0.	0.
EU=155	0.	0.	1.00E+02	0.	0.	0.	0.	0.	0.	0.	0.
DTMR	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	0.	0.	3.68E+10	3.39E+06	1.22E+06	1.19E+06	1.17E+06	1.02E+06	8.43E+05	5.90E+05	5.55E+05
URANIUM ORE INDPX	0.	0.	4.23E+02	3.49E+02	1.00E+02	1.37E+02	1.74E+02	1.17E+02	1.02E+02	6.76E+03	6.38E+03

A. VALUES LESS THAN 1.0E+10 HAVE BEEN DESIGNATED AS ZERO.

TABLE A.4.2b. Hazard Index--Once-Through Cycle--Growth Case 2, m³ water/MTHM(A)

RADIONUCLIDES (R)	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
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CM=245	0.	0.	4.18E+04	4.04E+04	3.47E+04	2.77E+04	1.42E+04	6.15E+02	9.49E+00	0.	0.
CM=246	0.	0.	1.09E+07	1.03E+00	9.42E-09	0.	0.	0.	0.	0.	0.
CM=243	0.	0.	1.45E+09	2.95E+01	4.45E-04	0.	0.	0.	0.	0.	0.
CM=242	0.	0.	3.06E+05	4.00E+04	4.01E+03	5.08E-05	0.	0.	0.	0.	0.
AM=243+AM=239	0.	0.	3.42E+06	3.90E+06	3.15E+06	2.20E+06	1.40E+06	1.72E+04	4.01E+02	0.	0.
AM=242+AM=242	0.	0.	1.94E+06	3.04E+05	3.11E+04	3.72E+04	0.	0.	0.	0.	0.
AM=241	0.	0.	8.05E+08	4.58E+06	2.06E+08	3.69E+05	1.44E+04	4.36E+02	4.60E+00	0.	0.
PU=242	0.	0.	3.09E+05	3.09E+05	3.09E+05	3.06E+05	1.03E+05	1.42E+05	2.87E+05	1.24E+05	4.96E+04
PU=241	0.	0.	1.81E+07	9.08E+02	7.76E+02	5.94E+02	1.45E+02	1.57E+01	1.92E+01	0.	0.
PU=240	0.	0.	8.82E+07	8.47E+07	8.04E+07	5.34E+07	1.20E+07	1.59E+05	3.14E+03	0.	0.
PU=239	0.	0.	5.73E+07	5.67E+07	5.49E+07	5.01E+07	4.47E+07	1.02E+07	1.42E+06	3.97E+01	2.69E+05
PU=238	0.	0.	2.92E+08	1.02E+07	2.05E+05	5.67E+04	0.	0.	0.	0.	0.
PU=236	0.	0.	1.66E+02	0.	0.	0.	0.	0.	0.	0.	0.
NP=237+NA=233	0.	0.	1.26E+05	2.35E+05	3.11E+05	3.69E+05	1.49E+05	1.44E+05	3.99E+05	3.15E+05	2.68E+05
U=238+TH=234+ Pj=234	0.	0.	2.37E+04	2.37E+04	2.37E+04	2.37E+04	2.37E+04	2.37E+04	2.37E+04	2.37E+04	2.37E+04
U=236	0.	0.	7.38E+03	7.34E+03	7.74E+03	9.01E+03	1.00E+04	1.15E+04	1.19E+04	1.14E+04	1.12E+04
U=235+TH=231	0.	0.	6.74E+02	6.78E+02	6.84E+02	7.23E+02	7.67E+02	9.44E+02	1.04E+03	1.06E+03	1.06E+03
U=234	0.	0.	4.28E+04	5.54E+04	5.47E+04	5.62E+04	4.46E+04	5.08E+04	4.55E+04	2.19E+04	1.33E+04
U=233	0.	0.	5.40E+00	1.52E+01	9.07E+01	6.82E+02	1.43E+03	6.79E+03	1.22E+04	2.84E+04	2.75E+04
U=232	0.	0.	3.12E+03	6.25E+01	5.04E+01	0.	0.	0.	0.	0.	0.
PA=231	0.	0.	5.93E+01	2.17E+02	4.70E+02	2.05E+03	4.03E+03	1.46E+04	2.49E+04	3.06E+04	3.05E+04
TH=230	0.	0.	3.79E+02	2.94E+03	6.73E+03	3.53E+04	4.95E+04	2.42E+05	4.36E+05	3.95E+05	2.20E+05
TH=229+ DAUGHTERS	0.	0.	7.97E+01	2.12E+01	1.12E+02	3.58E+03	1.08E+04	1.44E+05	3.35E+05	8.56E+05	8.31E+05
TH=228+ DAUGHTERS	0.	0.	6.64E+03	1.34E+02	1.10E+00	4.19E-02	9.10E-02	5.08E-01	1.14E+00	5.82E+00	1.16E+01
AC=227+ DAUGHTERS	0.	0.	7.87E+01	3.40E+02	7.49E+02	3.47E+03	7.21E+03	2.07E+04	4.46E+04	5.46E+04	5.46E+04
TH=232+ DAUGHTERS	0.	0.	2.10E-02	1.66E-01	3.64E-01	2.04E+00	6.04E+00	2.47E+01	5.55E+01	2.84E+02	5.66E+02
RA=226+ DAUGHTERS	0.	0.	4.11E+02	1.72E+04	7.99E+04	1.39E+06	3.43E+06	1.91E+07	2.96E+07	2.66E+07	1.88E+07
PR=216+ DAUGHTERS	0.	0.	7.01E+01	5.64E+03	2.78E+04	4.75E+05	1.24E+06	6.09E+06	1.01E+07	9.06E+06	5.04E+06
TOTAL	0.	0.	1.26E+09	6.14E+08	3.86E+08	1.09E+08	4.59E+07	4.15E+07	4.46E+07	3.75E+07	2.14E+07
URANIUM ORE INDEX	0.	0.	1.49E+01	7.05E+00	3.48E+00	1.25E+00	9.42E-01	4.77E-01	5.13E-01	4.31E-01	2.46E-01

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.
 B. TH=229, 7 DAUGHTERS ARE RA=225, AC=225, FR=221, AT=217, RI=213, PB=209 AND TL=204 IS 91% OF TH=229 AND PO=213 IS 91% OF TH=229.
 TH=228, 6 DAUGHTERS ARE RA=224, RN=220, PO=216, PB=212, AT=212 AND TL=208 IS 74% OF TH=228 AND PO=212 IS 64% OF TH=228.
 AC=227, 7 DAUGHTERS ARE TH=227, PA=223, RN=210, PO=215, PA=211, BI=211 AND TL=207.
 TH=232, 2 DAUGHTERS ARE PA=228 AND AC=228.
 PA=226, 5 DAUGHTERS ARE RN=222, PO=214, PB=214, BI=214 AND PH=214.
 PB=210, 2 DAUGHTERS ARE AT=210 AND PO=210.

NOTE. IN ACCOUNTING FOR THE ACTIVITY IN THIS MANNER, BRANCHING DECAY IN THE CASE OF TH=204 (36%) = PO=212 (64%), AND TL=209 (9%) = PO=221 (91%) WERE COUNTED AS A SINGLE DAUGHTER IN EACH CASE. MINOR BRANCHING (1% OR LESS) WAS IGNORED.

TABLE A.4.3a. Hazard Index--Once-Through Cycle--Growth Case 3, m³ water/MTHM(A)

Fission and Activation Products

MAJOR RADYONUCLIDES	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)								
	2000	2050	2070	500	1000	3000	10000	50000	100000	500000	1000000	
H-3	0.	0.	5.40E+03	6.47E-07	0.	0.	0.	0.	0.	0.	0.	0.
C-14	0.	0.	1.04E+03	9.88E+02	9.70E+02	9.73E+02	9.13E+02	2.69E+00	9.40E+03	0.	0.	0.
MN-54	0.	0.	2.70E-07	0.	0.	0.	0.	0.	0.	0.	0.	0.
FE-55	0.	0.	6.41E+01	0.	0.	0.	0.	0.	0.	0.	0.	0.
CO-60	0.	0.	1.27E+05	0.	0.	0.	0.	0.	0.	0.	0.	0.
NI-59	0.	0.	1.51E+04	1.51E+04	1.40E+04	1.45E+04	1.49E+04	9.41E+03	6.36E+03	1.99E+02	2.63E+00	0.
NI-63	0.	0.	1.01E+07	4.75E+05	1.10E+04	9.17E+10	0.	0.	0.	0.	0.	0.
SE-79	0.	0.	1.15E+05	1.15E+05	1.14E+05	1.09E+04	1.03E+04	4.76E+04	3.07E+04	5.59E+02	2.71E+00	0.
KR-85	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
RB-87	0.	0.	1.71E+01	1.71E+01	1.71E+01	1.71E+01	1.71E+01	1.71E+01	1.71E+01	1.71E+01	1.71E+01	1.71E+01
BR-90+Y-90	0.	0.	5.09E+10	2.29E+06	1.02E+01	0.	0.	0.	0.	0.	0.	0.
ZR-93	0.	0.	2.05E+03	2.05E+03	2.05E+03	2.05E+03	2.04E+03	2.00E+03	1.96E+03	1.63E+03	1.29E+03	0.
NO-23M	0.	0.	3.00E+03	4.10E+03	4.10E+03	4.10E+03	4.09E+03	4.01E+03	3.92E+03	3.26E+03	2.59E+03	0.
TC-99	0.	0.	4.23E+04	4.22E+04	4.22E+04	4.16E+04	4.09E+04	3.99E+04	3.94E+04	8.13E+03	1.56E+03	0.
RU-106+RH-106	0.	0.	2.21E+01	0.	0.	0.	0.	0.	0.	0.	0.	0.
PD-107	0.	0.	3.21E+04	3.21E+04	3.21E+04	3.21E+04	3.21E+04	3.20E+04	3.16E+04	3.06E+04	2.91E+04	0.
AG-110M	0.	0.	1.99E+08	0.	0.	0.	0.	0.	0.	0.	0.	0.
CD-113M	0.	0.	1.61E+06	3.04E+03	0.	0.	0.	0.	0.	0.	0.	0.
SB-125+Y-125M	0.	0.	7.19E+02	0.	0.	0.	0.	0.	0.	0.	0.	0.
SN-126+SB-126	0.	0.	4.67E+05	4.66E+05	4.44E+05	4.51E+05	4.76E+05	3.71E+05	2.34E+05	1.46E+04	4.59E+02	0.
I-129	0.	0.	5.44E+05	5.44E+05	5.44E+05	5.44E+05	4.44E+05	4.23E+05	5.42E+05	5.33E+05	5.22E+05	0.
CS-134	0.	0.	1.03E+06	0.	0.	0.	0.	0.	0.	0.	0.	0.
CS-135	0.	0.	2.64E+03	2.64E+03	2.64E+03	2.64E+03	2.63E+03	2.61E+03	2.58E+03	2.35E+03	2.10E+03	0.
CS-137+AA-137	0.	0.	1.13E+09	9.58E+04	9.58E+01	0.	0.	0.	0.	0.	0.	0.
CE-144+PR-144	0.	0.	9.19E+04	0.	0.	0.	0.	0.	0.	0.	0.	0.
PM-147	0.	0.	4.22E+03	0.	0.	0.	0.	0.	0.	0.	0.	0.
BM-151	0.	0.	1.75E+06	6.91E+04	1.29E+03	1.90E-11	0.	0.	0.	0.	0.	0.
EU-152	0.	0.	6.11E+03	4.05E-07	0.	0.	0.	0.	0.	0.	0.	0.
EU-154	0.	0.	2.27E+07	3.27E+01	2.09E-10	0.	0.	0.	0.	0.	0.	0.
EU-155	0.	0.	3.49E+01	0.	0.	0.	0.	0.	0.	0.	0.	0.
DTMFR	0.	0.	1.44E+10	0.	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	0.	0.	5.21E+10	4.16E+06	1.23E+06	1.20E+06	1.18E+06	1.03E+06	9.92E+05	5.94E+05	5.59E+05	0.
URANIUM ORE INDEX	0.	0.	5.99E+02	4.78E-02	1.02E+02	1.38E-02	1.76E+02	1.16E+02	1.03E+02	6.83E-03	6.43E-03	0.

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.

TABLE A.4.3b. Hazard Index--Once-Through Cycle--Growth Case 3, m³ water/MTHM(A)

Actinides

RADIONUCLIDES (B)	YEAR			GEOLOGICAL TIME (YEARS) BEYOND 1975							
	2040	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
CM=205	0.	0.	4.27E+04	4.12E+04	3.05E+04	2.83E+04	1.86E+04	6.29E+02	9.79E+00	0.	0.
CM=204	0.	0.	1.91E+07	3.37E+00	1.63E+08	0.	0.	0.	0.	0.	0.
CM=203	0.	0.	2.11E+05	3.19E+01	6.41E+04	0.	0.	0.	0.	0.	0.
CM=202	0.	0.	3.27E+05	5.13E+04	5.24E+03	6.28E+05	0.	0.	0.	0.	0.
AM=203+PM=239	0.	0.	3.47E+06	3.35E+06	3.20E+06	2.23E+06	1.41E+06	1.77E+04	4.06E+02	0.	0.
AM=202+PM=207	0.	0.	2.07E+06	3.25E+05	3.42E+04	3.97E+04	0.	0.	0.	0.	0.
AM=201	0.	0.	8.01E+08	4.72E+08	2.12E+08	3.41E+05	1.44E+00	5.40E+02	9.41E+00	0.	0.
PU=202	0.	0.	3.12E+05	3.12E+05	3.11E+05	3.09E+05	1.06E+05	2.45E+05	2.40E+05	1.25E+05	5.01E+04
PU=201	0.	0.	3.60E+07	8.26E+02	7.02E+02	5.66E+02	1.72E+02	1.70E+01	1.94E+01	0.	0.
PU=200	0.	0.	8.48E+07	8.42E+07	8.10E+07	5.37E+07	1.22E+07	5.42E+05	3.16E+03	0.	0.
PU=239	0.	0.	5.77E+07	5.71E+07	5.43E+07	5.05E+07	4.00E+07	1.23E+07	1.45E+06	4.00E+01	2.71E+05
PU=238	0.	0.	2.60E+06	1.13E+07	2.72E+05	6.06E+04	0.	0.	0.	0.	0.
PU=236	0.	0.	1.06E+00	0.	0.	0.	0.	0.	0.	0.	0.
NP=237+PA=233	0.	0.	1.23E+05	2.32E+05	3.12E+05	3.73E+05	3.73E+05	3.48E+05	3.62E+05	3.16E+05	2.71E+05
U=238+Th=234+ P=234M	0.	0.	2.37E+04	2.37E+04	2.47E+04	2.37E+04	2.47E+04	2.47E+04	2.37E+04	2.37E+04	2.37E+04
U=236	0.	0.	7.03E+03	7.59E+03	7.80E+03	9.07E+03	1.01E+04	1.16E+04	1.16E+04	1.15E+04	1.13E+04
U=235+Th=231	0.	0.	6.79E+02	6.83E+02	6.88E+02	7.28E+02	7.72E+02	9.71E+02	1.00E+03	1.06E+03	1.06E+03
U=234	0.	0.	4.16E+04	5.54E+04	5.71E+04	5.67E+04	4.40E+04	5.12E+04	4.59E+04	2.20E+04	1.34E+04
U=233	0.	0.	5.19E+00	3.34E+01	9.45E+01	6.85E+02	1.44E+03	6.45E+03	1.23E+04	2.07E+04	2.78E+04
U=232	0.	0.	3.59E+03	7.20E+01	5.44E+01	0.	0.	0.	0.	0.	0.
PA=231	0.	0.	5.01E+01	2.10E+02	4.26E+02	2.06E+03	4.05E+03	1.47E+04	2.91E+04	3.08E+04	3.08E+04
Th=230	0.	0.	3.11E+02	2.81E+03	6.46E+03	3.54E+04	4.09E+04	2.44E+05	4.39E+05	3.98E+05	2.21E+05
Th=229+7 DAUGHTERS	0.	0.	6.74E+01	1.94E+01	1.09E+02	3.48E+03	1.49E+04	1.45E+05	3.38E+05	8.65E+05	8.39E+05
Th=228+6 DAUGHTERS	0.	0.	7.69E+03	1.54E+02	1.26E+00	4.20E+02	9.16E+02	5.47E+01	1.15E+00	5.86E+00	1.17E+01
AC=227+7 DAUGHTERS	0.	0.	6.77E+01	3.64E+02	7.41E+02	3.67E+03	7.25E+03	2.99E+04	4.49E+04	5.50E+04	5.50E+04
Th=232+7 DAUGHTERS	0.	0.	1.73E+02	1.60E+01	3.41E+01	2.05E+00	4.07E+00	2.49E+01	5.99E+01	2.86E+02	5.70E+02
RA=226+5 DAUGHTERS	0.	0.	2.90E+02	1.47E+04	7.79E+04	1.39E+06	1.46E+06	1.02E+07	2.98E+07	2.69E+07	1.49E+07
PB=210+2 DAUGHTERS	0.	0.	4.57E+01	5.07E+03	2.45E+04	4.75E+05	1.24E+06	6.74E+06	1.02E+07	9.12E+06	5.06E+06
TOTAL	0.	0.	1.31E+09	6.30E+08	3.44E+08	1.10E+04	8.74E+07	4.18E+07	4.50E+07	3.78E+07	2.15E+07
URANIUM ORE INDEX	0.	0.	1.40E+01	7.24E+00	4.07E+00	1.26E+00	9.49E+01	4.41E+01	5.17E+01	4.34E+01	2.47E+01

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.
 B. TH=229, 7 DAUGHTERS ARE RA=225, AC=225, PM=221, AT=217, RI=213, PB=209 AND TL=200 +8 9% OF TH=229 AND PO=213 IS 91% OF TH=229.
 TH=228, 6 DAUGHTERS ARE RA=220, RN=227, PO=216, PB=212, RI=212 AND TL=208 IS 74% OF TH=228 AND PO=212 IS 64% OF TH=228.
 AC=227, 7 DAUGHTERS ARE TH=227, RA=223, RN=219, PO=215, PM=211, BI=211 AND TL=207.
 TH=232, 2 DAUGHTERS ARE RA=228 AND AC=228.
 RA=226, 5 DAUGHTERS ARE RN=222, PM=214, PB=214, BI=214 AND PM=214.
 PB=210, 2 DAUGHTERS ARE RI=210 AND PO=210.

NOTES. IN ACCOUNTING FOR THE ACTIVITY IN THIS MANNER, BRANCHING DECAY IN THE CASE OF TL=209 (34%) = PO=212 (64%), AND TL=209 (9%) = PO=211 (91%) WERE COUNTED AS A SINGLE DAUGHTER IN EACH CASE. MINOR BRANCHING (1% OR LESS) WAS IGNORED.

TABLE A.4.4a. Hazard Index--Once-Through Cycle--Growth Case 4, m³ water/MTHM(A)

Fission and Activation Products

MAJOR RADIONUCLIDES	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	300	1000	5000	10000	50000	100000	500000	1000000
H=3	0.	0.	8.02E+03	9.42E-07	0.	0.	0.	0.	0.	0.	0.
C=14	0.	0.	1.04E+03	9.89E+02	9.41E+02	5.74E+02	3.14E+02	2.49E+00	5.90E+03	0.	0.
MN=54	0.	0.	1.63E+06	0.	0.	0.	0.	0.	0.	0.	0.
FE=55	0.	0.	2.35E+02	0.	0.	0.	0.	0.	0.	0.	0.
CO=60	0.	0.	2.93E+09	0.	0.	0.	0.	0.	0.	0.	0.
NI=59	0.	0.	1.51E+04	1.50E+04	1.09E+04	1.44E+04	7.48E+04	9.78E+03	6.34E+03	1.99E+02	2.62E+00
NI=63	0.	0.	1.05E+07	4.95E+05	1.15E+04	9.56E+10	0.	0.	0.	0.	0.
SE=79	0.	0.	1.14E+05	1.14E+05	1.13E+03	1.08E+05	1.03E+05	6.71E+04	3.94E+04	5.55E+02	2.69E+00
KR=85	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
RB=87	0.	0.	1.69E+01	1.69E+01	1.49E+01	1.69E+01	7.49E+01	7.49E+01	1.69E+01	1.69E+01	1.69E+01
BR=90+Y=90	0.	0.	5.89E+10	2.65E+06	1.18E+01	0.	0.	0.	0.	0.	0.
ZR=93	0.	0.	2.04E+03	2.03E+03	2.03E+03	2.03E+03	2.03E+03	1.99E+03	1.94E+03	1.62E+03	1.28E+03
NR=93M	0.	0.	3.79E+03	4.07E+03	4.07E+03	4.07E+03	4.06E+03	3.98E+03	3.89E+03	3.23E+03	2.57E+03
TC=99	0.	0.	4.20E+04	4.20E+04	4.19E+04	4.13E+04	4.07E+04	3.96E+04	3.02E+04	6.08E+03	1.55E+03
RU=106+PM=106	0.	0.	1.35E+00	0.	0.	0.	0.	0.	0.	0.	0.
PO=107	0.	0.	3.20E+04	3.20E+04	3.20E+04	3.20E+04	3.20E+04	3.19E+04	3.17E+04	3.05E+04	2.90E+04
AG=110M	0.	0.	1.66E-07	0.	0.	0.	0.	0.	0.	0.	0.
CD=113M	0.	0.	2.25E+06	4.25E+03	0.	0.	0.	0.	0.	0.	0.
BB=125+Y=125M	0.	0.	2.69E+03	0.	0.	0.	0.	0.	0.	0.	0.
BN=126+BB=126	0.	0.	4.65E+05	4.66E+05	4.62E+05	4.49E+05	4.34E+05	3.59E+05	2.33E+05	1.46E+04	4.57E+02
I=129	0.	0.	5.41E+05	5.41E+05	5.41E+05	5.41E+05	5.40E+05	5.40E+05	5.38E+05	5.30E+05	5.19E+05
CS=134	0.	0.	4.78E+06	0.	0.	0.	0.	0.	0.	0.	0.
CS=135	0.	0.	2.61E+03	2.61E+03	2.61E+03	2.61E+03	2.60E+03	2.48E+03	2.55E+03	2.33E+03	2.07E+03
CS=137+PA=137	0.	0.	1.30E+09	1.10E+05	1.06E+00	0.	0.	0.	0.	0.	0.
CE=144+PR=144	0.	0.	5.03E+03	0.	0.	0.	0.	0.	0.	0.	0.
PH=147	0.	0.	1.38E+04	0.	0.	0.	0.	0.	0.	0.	0.
SM=151	0.	0.	1.83E+06	7.20E+04	1.74E+03	1.98E+11	0.	0.	0.	0.	0.
EU=152	0.	0.	8.86E+03	5.88E+07	0.	0.	0.	0.	0.	0.	0.
EU=154	0.	0.	3.01E+07	7.01E+01	2.78E+10	0.	0.	0.	0.	0.	0.
EU=155	0.	0.	1.72E+02	0.	0.	0.	0.	0.	0.	0.	0.
OTHFR	0.	0.	6.35E+10	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	0.	0.	6.03E+10	4.55E+06	1.23E+06	1.20E+06	1.17E+06	1.02E+06	8.07E+05	5.91E+05	5.56E+05
URANIUM ORE INDEX	0.	0.	6.93E+02	5.23E+02	1.01E+02	1.37E+02	1.35E+02	1.17E+02	1.02E+02	6.79E+03	6.39E+03

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.

TABLE A.4.4b. Hazard: Index--Once-Through Cycle--Growth Case 4, m³ water/MTHM(A)

Actinides

RADIONUCLIDES (N)	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
CM=245	0.	0.	4.23E+04	4.08E+04	3.02E+04	2.80E+04	1.44E+04	6.63E+02	9.70E+00	0.	0.
CM=244	0.	0.	2.04E+07	4.32E+00	2.08E+08	0.	0.	0.	0.	0.	0.
CM=243	0.	0.	2.39E+05	3.62E+01	7.15E+04	0.	0.	0.	0.	0.	0.
CM=242	0.	0.	3.33E+05	5.22E+04	5.74E+03	6.40E+05	0.	0.	0.	0.	0.
AM=243+MP=239	0.	0.	3.05E+06	3.32E+06	3.17E+06	2.21E+06	1.00E+06	3.74E+04	4.03E+02	0.	0.
AM=242+AM=242	0.	0.	2.11E+06	3.31E+05	3.78E+04	4.04E+04	0.	0.	0.	0.	0.
AM=241	0.	0.	6.23E+06	4.75E+08	2.14E+08	3.83E+05	1.86E+04	6.04E+02	9.72E+00	0.	0.
PU=242	0.	0.	3.10E+05	3.10E+05	3.09E+05	3.07E+05	1.04E+05	2.83E+05	2.58E+05	1.24E+05	4.98E+04
PU=241	0.	0.	4.96E+07	5.18E+02	7.45E+02	5.61E+02	3.49E+02	1.29E+01	1.94E+01	0.	0.
PU=240	0.	0.	8.48E+07	6.52E+07	1.10E+07	5.37E+07	1.22E+07	5.72E+05	3.16E+03	0.	0.
PU=239	0.	0.	5.78E+07	5.71E+07	5.44E+07	5.05E+07	4.00E+07	1.23E+07	3.45E+06	4.01E+01	2.71E+05
PU=238	0.	0.	2.70E+08	1.18E+07	2.81E+05	6.17E+04	0.	0.	0.	0.	0.
PU=236	0.	0.	5.64E+00	0.	0.	0.	0.	0.	0.	0.	0.
NP=237+PA=233	0.	0.	1.20E+05	2.27E+05	3.10E+05	3.71E+05	1.71E+05	3.47E+05	3.61E+05	3.17E+05	2.70E+05
U=236+TM=234+ P=234M	0.	0.	2.37E+04	2.37E+04	2.47E+04	2.37E+04	2.47E+04	2.47E+04	2.37E+04	2.37E+04	2.37E+04
U=236	0.	0.	7.36E+03	7.52E+03	7.73E+03	9.00E+03	1.00E+04	1.15E+04	1.14E+04	1.14E+04	1.12E+04
U=235+TM=231	0.	0.	6.68E+02	6.78E+02	6.77E+02	7.17E+02	7.41E+02	9.40E+02	1.03E+03	1.05E+03	1.05E+03
U=234	0.	0.	4.01E+04	5.03E+04	5.43E+04	5.58E+04	5.42E+04	4.04E+04	4.52E+04	2.18E+04	1.33E+04
U=233	0.	0.	4.87E+00	3.20E+01	9.19E+01	6.80E+02	1.03E+03	6.42E+03	1.23E+04	2.86E+04	2.77E+04
U=232	0.	0.	3.76E+03	7.94E+01	6.12E+01	0.	0.	0.	0.	0.	0.
PA=231	0.	0.	5.11E+01	2.02E+02	4.16E+02	2.02E+03	3.99E+03	1.45E+04	2.48E+04	3.05E+04	3.04E+04
TM=230	0.	0.	2.78E+02	2.67E+03	6.09E+03	3.47E+04	4.88E+04	2.40E+05	4.33E+05	3.93E+05	2.19E+05
TM=229+ DAUGHTERS	0.	0.	5.16E+01	1.82E+01	1.06E+02	3.84E+03	1.08E+04	1.45E+05	3.37E+05	8.62E+05	8.37E+05
TM=228+ DAUGHTERS	0.	0.	8.06E+03	1.62E+02	1.42E+00	4.15E+02	9.08E+02	5.08E+01	1.14E+00	5.83E+00	1.16E+01
AC=227+ DAUGHTERS	0.	0.	6.06E+01	3.08E+02	7.04E+02	3.61E+03	7.03E+03	2.95E+04	4.44E+04	5.45E+04	5.44E+04
TM=232+ DAUGHTERS	0.	0.	1.38E+02	1.52E+01	3.44E+01	2.02E+00	4.03E+00	2.47E+01	5.45E+01	2.84E+02	5.67E+02
RA=226+ DAUGHTERS	0.	0.	2.05E+02	1.46E+04	7.43E+04	1.37E+06	3.60E+06	1.49E+07	2.94E+07	2.65E+07	1.08E+07
PB=210+ DAUGHTERS	0.	0.	3.76E+01	4.64E+03	2.46E+04	4.69E+05	1.22E+06	6.24E+06	1.00E+07	9.02E+06	5.03E+06
TOTAL	0.	0.	1.32E+09	6.34E+08	3.55E+08	1.10E+08	4.33E+07	4.74E+07	4.44E+07	3.74E+07	2.13E+07
URANIUM ORE INDEX	0.	0.	1.92E+01	7.28E+00	4.08E+00	1.26E+00	9.48E+01	4.76E+01	5.10E+01	4.30E+01	2.45E+01

A. VALUES LESS THAN 1.0E+10 HAVE BEEN DESIGNATED AS ZERO.
 B. TM=229, 7 DAUGHTERS ARE RA=225, AC=225, FR=221, AT=217, RT=213, PB=209 AND TL=200 IS 9% OF TM=229 AND PD=213 IS 91% OF TM=229.
 TM=228, 6 DAUGHTERS ARE RA=224, RN=220, PD=216, PB=212, BI=212 AND TL=208 IS 34% OF TM=228 AND PD=212 IS 64% OF TM=228.
 AC=227, 7 DAUGHTERS ARE TM=227, RA=223, RN=219, PD=215, PR=211, AI=211 AND TL=207.
 TM=232, 2 DAUGHTERS ARE RA=228 AND AC=228.
 PA=226, 5 DAUGHTERS ARE RN=222, PD=214, PB=214, BI=214 AND PD=214.
 PB=210, 2 DAUGHTERS ARE BI=210 AND PD=210.

NOTE. IN ACCOUNTING FOR THE ACTIVITY IN THIS MANNER, BRANCHING DECAY IN THE CASE OF TL=208 (36%) = PD=212 (64%), AND TL=209 (9%) = PD=211 (91%) WERE COUNTED AS A SINGLE DAUGHTER IN EACH CASE. MINOR BRANCHING (1% OR LESS) WAS IGNORED.

TABLE A.4.5a. Hazard Index--Once-Through Cycle--Growth Case 5, m³ water/MTHM(A)

Fission and Activation Products

MAJOR RAOYDNUCLIDES	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
M=3	0.	0.	9.56E+03	1.12E+06	0.	0.	0.	0.	0.	0.	0.
C=10	0.	0.	1.05E+03	9.97E+02	9.79E+02	5.79E+02	7.16E+02	2.41E+00	5.96E+03	0.	0.
MN=94	0.	0.	2.22E+06	0.	0.	0.	0.	0.	0.	0.	0.
FE=95	0.	0.	3.19E+02	0.	0.	0.	0.	0.	0.	0.	0.
CO=60	0.	0.	3.83E+05	0.	0.	0.	0.	0.	0.	0.	0.
NI=89	0.	0.	1.52E+04	1.51E+04	1.40E+04	1.45E+04	1.39E+04	9.44E+03	6.38E+03	2.00E+02	2.63E+00
NI=63	0.	0.	1.09E+07	5.13E+05	1.19E+04	9.89E+10	0.	0.	0.	0.	0.
SE=79	0.	0.	1.15E+05	1.14E+05	1.16E+05	1.09E+05	1.03E+05	6.74E+04	3.96E+04	5.57E+02	2.70E+00
KR=85	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
RB=87	0.	0.	1.70E+01	1.70E+01	1.70E+01	1.70E+01	1.70E+01	1.70E+01	1.70E+01	1.70E+01	1.70E+01
SR=90+Y=90	0.	0.	6.44E+10	2.90E+06	1.79E+01	0.	0.	0.	0.	0.	0.
ZR=93	0.	0.	2.04E+03	2.04E+03	2.04E+03	2.04E+03	2.04E+03	2.00E+03	1.95E+03	1.62E+03	1.29E+03
NB=93M	0.	0.	3.76E+03	4.09E+03	4.09E+03	4.08E+03	4.07E+03	4.00E+03	3.91E+03	3.25E+03	2.58E+03
TC=99	0.	0.	4.22E+04	4.22E+04	4.21E+04	4.15E+04	4.09E+04	3.88E+04	3.04E+04	8.12E+03	1.56E+03
RU=106+PM=106	0.	0.	1.87E+00	0.	0.	0.	0.	0.	0.	0.	0.
PO=107	0.	0.	3.23E+04	3.22E+04	3.22E+04	3.22E+04	3.22E+04	3.21E+04	3.19E+04	3.07E+04	2.92E+04
AG=110M	0.	0.	2.34E+07	0.	0.	0.	0.	0.	0.	0.	0.
CO=113M	0.	0.	2.66E+06	5.03E+03	0.	0.	0.	0.	0.	0.	0.
BB=125+TE=125M	0.	0.	3.68E+03	0.	0.	0.	0.	0.	0.	0.	0.
BN=126+SB=126	0.	0.	4.68E+05	4.66E+05	4.65E+05	4.52E+05	4.47E+05	3.41E+05	2.34E+05	1.47E+04	4.60E+02
I=129	0.	0.	5.44E+05	5.44E+05	5.44E+05	5.43E+05	5.43E+05	5.42E+05	5.41E+05	5.33E+05	5.22E+05
CB=134	0.	0.	6.65E+06	0.	0.	0.	0.	0.	0.	0.	0.
CB=135	0.	0.	2.62E+03	2.62E+03	2.62E+03	2.62E+03	2.61E+03	2.59E+03	2.56E+03	2.33E+03	2.08E+03
CB=137+BA=137	0.	0.	1.01E+09	1.20E+05	1.16E+00	0.	0.	0.	0.	0.	0.
CE=144+PR=144	0.	0.	6.90E+03	0.	0.	0.	0.	0.	0.	0.	0.
PM=147	0.	0.	1.85E+04	0.	0.	0.	0.	0.	0.	0.	0.
SM=151	0.	0.	1.89E+06	7.45E+04	1.79E+03	2.05E+11	0.	0.	0.	0.	0.
EU=152	0.	0.	1.06E+04	7.00E+07	0.	0.	0.	0.	0.	0.	0.
EU=154	0.	0.	3.50E+07	8.14E+01	3.23E+10	0.	0.	0.	0.	0.	0.
EU=155	0.	0.	2.40E+02	0.	0.	0.	0.	0.	0.	0.	0.
DTMFR	0.	0.	1.15E+09	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	0.	0.	6.59E+10	4.83E+06	1.23E+06	1.20E+06	1.18E+06	1.03E+06	8.92E+05	5.94E+05	5.59E+05
URANIUM ONE INDEX	0.	0.	7.58E+02	5.55E+02	1.02E+02	1.38E+02	1.36E+02	1.18E+02	1.03E+02	6.83E+03	6.43E+03

A. VALUES LESS THAN 1.0E+10 HAVE BEEN DESIGNATED AS ZERO.

TABLE A.4.5b. Hazard Index--Once-Through Cycle--Growth Case 5, m³ water/MTHM(A)

Actinides

RADIONUCLIDES (B)	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
CM=245	0.	0.	4.26E+04	4.12E+04	3.05E+04	2.82E+04	1.46E+04	6.08E+02	9.78E+00	0.	0.
CM=244	0.	0.	2.81E+07	4.97E+00	2.40E+08	0.	0.	0.	0.	0.	0.
CM=243	0.	0.	2.60E+05	3.94E+01	7.79E+04	0.	0.	0.	0.	0.	0.
CM=242	0.	0.	3.41E+05	5.35E+04	5.07E+03	6.55E+05	0.	0.	0.	0.	0.
AM=243+AM=239	0.	0.	3.47E+06	3.35E+06	3.20E+06	2.23E+06	1.42E+06	3.77E+04	4.07E+02	0.	0.
AM=242+AM=242	0.	0.	2.16E+06	3.39E+05	3.07E+04	4.14E+04	0.	0.	0.	0.	0.
AM=241	0.	0.	8.21E+08	4.82E+08	2.17E+08	3.88E+05	1.47E+04	6.09E+02	9.40E+00	0.	0.
PU=242	0.	0.	3.12E+05	3.12E+05	3.12E+05	3.10E+05	7.07E+05	2.45E+05	2.60E+05	1.25E+05	5.02E+04
PU=241	0.	0.	5.40E+07	8.25E+02	7.92E+02	5.66E+02	7.72E+02	1.70E+01	1.96E+01	0.	0.
PU=240	0.	0.	8.94E+07	8.98E+07	8.15E+07	5.41E+07	3.24E+07	5.76E+05	3.18E+03	0.	0.
PU=239	0.	0.	5.40E+07	5.74E+07	5.46E+07	5.08E+07	4.02E+07	1.03E+07	3.07E+06	4.02E+01	2.72E+05
PU=238	0.	0.	2.40E+08	1.22E+07	2.90E+05	6.32E+04	0.	0.	0.	0.	0.
PU=236	0.	0.	7.80E+00	0.	0.	0.	0.	0.	0.	0.	0.
NP=237+PA=233	0.	0.	1.19E+05	2.27E+05	3.11E+05	3.74E+05	3.74E+05	3.49E+05	3.64E+05	3.19E+05	2.72E+05
U=238+TH=234+ PA=238M	0.	0.	2.37E+04	2.37E+04	2.47E+04	2.37E+04	2.47E+04	2.47E+04	2.38E+04	2.38E+04	2.38E+04
U=236	0.	0.	7.38E+03	7.54E+03	7.75E+03	9.03E+03	1.01E+04	1.16E+04	1.16E+04	1.14E+04	1.13E+04
U=235+TH=231	0.	0.	6.61E+02	6.64E+02	6.70E+02	7.10E+02	7.45E+02	9.44E+02	1.03E+03	1.05E+03	1.05E+03
U=234	0.	0.	3.95E+04	5.42E+04	5.42E+04	5.58E+04	7.41E+04	5.00E+04	4.52E+04	2.18E+04	1.33E+04
U=233	0.	0.	4.70E+00	3.16E+01	9.19E+01	6.84E+02	1.04E+03	6.47E+03	1.24E+04	2.88E+04	2.99E+04
U=232	0.	0.	3.92E+03	7.87E+01	6.78E+01	0.	0.	0.	0.	0.	0.
PA=231	0.	0.	4.96E+01	1.98E+02	4.10E+02	2.00E+03	7.05E+03	1.44E+04	2.47E+04	3.03E+04	3.03E+04
TH=230	0.	0.	2.48E+02	2.63E+03	6.45E+03	3.47E+04	4.87E+04	2.40E+05	4.33E+05	3.93E+05	2.20E+05
TH=229+7 DAUGHTERS	0.	0.	4.73E+01	1.78E+01	1.05E+02	3.86E+03	1.09E+04	1.46E+05	3.39E+05	8.68E+05	8.43E+05
TH=228+6 DAUGHTERS	0.	0.	8.41E+03	1.69E+02	1.78E+00	4.16E+02	9.10E+02	5.40E+01	1.14E+00	5.85E+00	1.17E+01
AC=227+7 DAUGHTERS	0.	0.	5.73E+01	3.41E+02	7.44E+02	3.57E+03	7.06E+03	2.93E+04	4.41E+04	5.42E+04	5.41E+04
TH=226+2 DAUGHTERS	0.	0.	1.28E+02	1.51E+01	3.53E+01	2.03E+00	4.04E+00	2.48E+01	5.58E+01	2.85E+02	5.69E+02
RA=226+5 DAUGHTERS	0.	0.	2.16E+02	1.42E+04	7.45E+04	1.36E+06	7.59E+06	1.49E+07	2.94E+07	2.65E+07	1.48E+07
PB=210+2 DAUGHTERS	0.	0.	3.22E+01	4.49E+03	2.44E+04	4.64E+05	1.22E+06	6.03E+06	1.00E+07	9.02E+06	5.04E+06
TOTAL	0.	0.	1.34E+09	6.42E+08	3.49E+08	1.10E+08	4.78E+07	4.15E+07	4.44E+07	3.74E+07	2.14E+07
URANIUM ORE INDEX	0.	0.	1.54E+01	7.37E+00	4.13E+00	1.27E+00	9.43E+01	0.77E+01	5.10E+01	4.30E+01	2.46E+01

A. VALUES LESS THAN 1.0E+10 HAVE BEEN DESIGNATED AS ZERO.
 B. TH=229, 7 DAUGHTERS ARE RA=225, AC=225, FR=221, AT=217, BI=213, PB=209 AND TL=209 IS 9% OF TH=229 AND PO=213 IS 91% OF TH=229.
 TH=228, 6 DAUGHTERS ARE RA=224, RN=220, PO=216, PB=212, BI=212 AND TL=208 IS 36% OF TH=228 AND PO=212 IS 64% OF TH=228.
 AC=227, 7 DAUGHTERS ARE TH=227, RA=223, RN=219, PO=215, PB=211, BI=211 AND TL=207.
 TH=222, 2 DAUGHTERS ARE RA=228 AND AC=228.
 PA=229, 5 DAUGHTERS ARE RN=222, PO=218, PB=214, BI=214 AND PN=214.
 PB=210, 2 DAUGHTERS ARE BI=210 AND PD=210.

NOTE. IN ACCOUNTING FOR THE ACTIVITY IN THIS MANNER, BRANCHING DECAY IN THE CASE OF TL=208 (36%) = PD=212 (64%), AND TL=209 (9%) = PD=213 (91%) WERE COUNTED AS A SINGLE DAUGHTER IN EACH CASE. MINOR BRANCHING (1% OR LESS) WAS IGNORED.

TABLE A.4.6a. Hazard Index--Reprocessing Cycle--Growth Case 3--1990 Reprocessing Startup, m³ water/MTHM(A)

Fission and Activation Products

MAJOR RADIONUCLIDES	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
H-3	0.	0.	8.80E+02	1.04E-07	0.	0.	0.	0.	0.	0.	0.
C-14	0.	0.	9.84E+02	9.42E+02	8.86E+02	5.46E+02	2.99E+02	2.17E+00	5.62E+03	0.	0.
MN-54	0.	0.	2.96E-07	0.	0.	0.	0.	0.	0.	0.	0.
FE-55	0.	0.	5.72E+01	0.	0.	0.	0.	0.	0.	0.	0.
CO-60	0.	0.	1.17E+05	0.	0.	0.	0.	0.	0.	0.	0.
NI-63	0.	0.	1.31E+04	1.31E+04	1.70E+04	1.26E+04	1.21E+04	8.92E+03	5.93E+03	1.73E+02	2.26E+00
NI-64	0.	0.	8.75E+06	4.10E+05	9.81E+03	7.92E+10	0.	0.	0.	0.	0.
SE-79	0.	0.	1.09E+05	1.09E+05	1.09E+05	1.04E+05	9.83E+04	8.62E+04	7.77E+04	5.31E+02	8.57E+00
KR-85	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
RB-87	0.	0.	1.51E-01	1.51E+01	1.91E-01	1.91E-01	1.91E-01	1.91E-01	1.51E+01	1.51E+01	1.51E+01
BR-90+Y-90	0.	0.	4.40E+10	2.00E+06	8.84E+00	0.	0.	0.	0.	0.	0.
ZR-93	0.	0.	2.73E+03	2.03E+03	2.03E+03	2.03E+03	2.03E+03	1.99E+03	1.99E+03	1.61E+03	1.26E+03
NB-93M	0.	0.	3.47E+03	4.06E+03	4.06E+03	4.06E+03	4.05E+03	4.07E+03	3.88E+03	3.23E+03	2.56E+03
TC-99	0.	0.	4.19E+04	4.18E+04	4.18E+04	4.12E+04	4.06E+04	4.05E+04	3.01E+04	8.06E+03	1.95E+03
HU-106+PM-106	0.	0.	2.56E-01	0.	0.	0.	0.	0.	0.	0.	0.
PD-107	0.	0.	4.01E+04	4.01E+04	4.01E+04	4.01E+04	4.00E+04	4.09E+04	3.97E+04	7.81E+04	3.63E+04
AG-110M	0.	0.	2.84E-08	0.	0.	0.	0.	0.	0.	0.	0.
CD-113M	0.	0.	3.27E+06	6.17E+03	0.	0.	0.	0.	0.	0.	0.
SB-125+TE-125M	0.	0.	8.70E+02	0.	0.	0.	0.	0.	0.	0.	0.
BN-126+RA-126	0.	0.	5.45E+05	5.45E+05	5.01E+05	5.26E+05	4.08E+05	4.45E+05	2.73E+05	1.71E+04	5.35E+02
I-129	0.	0.	5.80E+05	5.80E+05	5.80E+05	5.79E+05	4.79E+05	4.78E+05	5.77E+05	5.68E+05	5.56E+05
CS-134	0.	0.	1.03E+06	0.	0.	0.	0.	0.	0.	0.	0.
CS-135	0.	0.	3.72E+03	3.32E+03	3.72E+03	3.72E+03	4.41E+03	3.58E+03	3.25E+03	2.96E+03	2.64E+03
CS-137+RA-137	0.	0.	1.13E+09	9.55E+04	9.55E+01	0.	0.	0.	0.	0.	0.
CE-140+PR-140	0.	0.	8.96E-04	0.	0.	0.	0.	0.	0.	0.	0.
PM-147	0.	0.	4.11E+03	0.	0.	0.	0.	0.	0.	0.	0.
SM-151	0.	0.	1.47E+06	7.36E+04	1.47E+03	2.03E-11	0.	0.	0.	0.	0.
EU-152	0.	0.	1.06E+04	7.05E+07	0.	0.	0.	0.	0.	0.	0.
EU-154	0.	0.	2.74E+07	6.76E+01	2.43E+10	0.	0.	0.	0.	0.	0.
EU-155	0.	0.	4.15E+01	0.	0.	0.	0.	0.	0.	0.	0.
DTFR	0.	0.	1.31E+10	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	0.	0.	4.55E+10	3.91E+06	1.74E+06	1.31E+06	1.29E+06	1.12E+06	9.72E+05	6.40E+05	6.01E+05
URANIUM ORE INDEX	0.	0.	5.23E+02	4.50E+02	1.45E+02	1.91E+02	1.05E+02	1.59E+02	1.12E+02	7.35E+03	6.91E+03

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.

TABLE A.4.6b. Hazard Index--Reprocessing Cycle--Growth Case 3--1990 Reprocessing Startup, m³ water/MTHM(A)

Actinides

RADIONUCLIDES (A)	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
CM-245	0.	0.	7.51E+04	7.26E+05	6.04E+05	4.98E+04	4.57E+05	1.74E+04	1.72E+02	0.	0.
CM-244	0.	0.	2.15E+08	4.79E+01	1.43E-17	0.	0.	0.	0.	0.	0.
CM-243	0.	0.	5.09E+09	4.41E+01	1.44E-03	0.	0.	0.	0.	0.	0.
CM-242	0.	0.	2.46E+06	3.71E+05	3.79E+04	4.54E+04	0.	0.	0.	0.	0.
AM-243+PM-239	0.	0.	1.42E+07	1.56E+07	1.09E+07	1.04E+07	4.61E+05	1.76E+05	1.90E+03	0.	0.
AM-242+AM-242	0.	0.	1.40E+07	2.45E+06	2.00E+05	2.47E-03	0.	0.	0.	0.	0.
AM-241	0.	0.	4.44E+04	2.36E+08	1.06E+08	4.96E+05	4.24E+05	1.74E+04	1.73E+02	0.	0.
PU-242	0.	0.	1.09E+05	1.09E+05	1.09E+05	1.04E+04	1.07E+04	4.09E+04	4.11E+04	4.39E+04	1.76E+04
PU-241	0.	0.	9.48E+06	1.45E+04	1.49E+04	9.97E+03	4.45E+03	2.39E+02	3.45E+00	0.	0.
PU-240	0.	0.	1.03E+07	1.05E+07	1.48E+07	9.16E+06	4.04E+06	4.07E+04	5.38E+02	0.	0.
PU-239	0.	0.	3.01E+06	3.10E+06	3.23E+06	3.91E+06	4.20E+06	2.09E+06	5.09E+05	5.91E+00	4.00E+06
PU-238	0.	0.	4.20E+07	5.88E+06	4.14E+05	4.38E+04	0.	0.	0.	0.	0.
PU-236	0.	0.	6.94E+02	0.	0.	0.	0.	0.	0.	0.	0.
NP-237+PA-233	0.	0.	2.07E+05	2.59E+05	2.05E+05	3.26E+04	4.27E+04	4.34E+05	3.19E+05	2.80E+05	2.38E+05
U-238+Th-234+Pa-234m	0.	0.	1.96E+02	1.96E+02	1.96E+02	1.96E+02	1.96E+02	1.96E+02	1.96E+02	1.96E+02	1.96E+02
U-236	0.	0.	9.13E+01	1.17E+02	1.43E+02	3.70E+02	4.05E+02	7.09E+02	4.02E+02	7.93E+02	7.82E+02
U-235+Th-231	0.	0.	5.45E+00	5.56E+00	5.07E+00	4.46E+00	1.24E+01	4.72E+01	4.76E+01	5.10E+01	5.09E+01
U-234	0.	0.	2.44E+03	7.19E+03	8.15E+03	4.12E+03	4.01E+03	7.17E+03	4.24E+03	2.09E+03	5.40E+02
U-233	0.	0.	4.06E+00	3.99E+01	9.07E+01	6.21E+02	1.24E+03	4.04E+03	1.09E+04	2.53E+04	2.45E+04
U-232	0.	0.	3.06E+05	1.10E+04	8.09E+01	0.	0.	0.	0.	0.	0.
PA-231	0.	0.	3.64E+01	3.78E+01	3.02E+01	5.29E+01	7.04E+01	4.09E+02	1.01E+03	1.47E+03	1.47E+03
TH-230	0.	0.	4.44E+01	3.13E+02	4.43E+02	4.98E+03	9.04E+03	4.02E+04	4.13E+04	4.34E+04	1.22E+04
TH-229+ DAUGHTERS	0.	0.	3.44E+01	2.28E+01	1.22E+02	3.62E+03	1.44E+04	1.04E+05	2.98E+05	7.82E+05	7.39E+05
TH-228+ DAUGHTERS	0.	0.	1.17E+06	2.35E+04	1.01E+02	9.47E+04	2.47E+03	2.47E+02	5.21E+02	2.78E+01	5.56E+01
AC-227+ DAUGHTERS	0.	0.	5.00E+01	6.72E+01	7.01E+01	9.46E+01	1.00E+02	4.02E+02	1.41E+03	2.63E+03	2.63E+03
TH-232+ DAUGHTERS	0.	0.	2.04E+04	2.06E+03	5.20E+03	4.62E+02	1.40E+01	1.16E+00	2.50E+00	1.35E+01	2.71E+01
RA-226+ DAUGHTERS	0.	0.	7.03E+01	1.69E+03	9.47E+03	1.94E+05	4.18E+05	2.72E+06	4.16E+06	2.92E+06	8.23E+05
PB-210+ DAUGHTERS	0.	0.	1.39E+01	5.36E+02	3.26E+03	4.61E+04	1.76E+05	4.35E+05	1.02E+06	4.95E+05	2.80E+05
TOTAL	0.	0.	7.93E+08	2.79E+08	1.00E+08	2.54E+07	1.82E+07	4.40E+04	4.48E+06	5.08E+06	2.14E+06
URANIUM ORE INDEX	0.	0.	9.14E+00	3.21E+00	1.41E+00	2.92E-01	2.09E-01	7.09E-02	7.91E-02	5.84E-02	2.46E-02

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.
 B. TH-229, 7 DAUGHTERS ARE RA-225, AC-225, PH-221, AT-217, AT-213, PB-209 AND TL-206 IS 9% OF TH-229 AND DN-213 IS 9% OF TH-229.
 TH-228, 6 DAUGHTERS ARE RA-224, RN-220, PO-216, PH-212, PI-212 AND TL-208 IS 3% OF TH-228 AND PD-212 IS 6% OF TH-228.
 AC-227, 7 DAUGHTERS ARE TH-227, RA-223, RN-219, PO-215, PH-211, RI-211 AND TL-207.
 TH-232, 2 DAUGHTERS ARE RA-228 AND AC-228.
 PA-226, 5 DAUGHTERS ARE RN-222, PD-214, PB-214, BI-214 AND PH-214.
 PB-210, 2 DAUGHTERS ARE RI-210 AND PO-210.

NOTE. IN ACCOUNTING FOR THE ACTIVITY IN THIS MANNER, BRANCHING DECAY IN THE CASE OF TI-204 (3%) = PO-212 (6%), AND TL-209 (9%) = PO-231 (9%) WERE COUNTED AS A SINGLE DAUGHTER IN EACH CASE. MINOR BRANCHING (1% OR LESS) WAS IGNORED.

TABLE A.4.7a. Hazard Index--Reprocessing Cycle--Growth Case 3--2010 Reprocessing Startup, m³ water/MTHM(A)

Fission and Activation Products

MAJOR RADIONUCLIDES	YEAR			GENOTOXIC TYPE (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
H-3	0.	0.	8.54E+02	1.01E+07	0.	0.	0.	0.	0.	0.	0.
C-14	0.	0.	1.10E+03	1.05E+03	9.00E+02	6.11E+02	3.34E+02	2.49E+00	4.28E+03	0.	0.
MN-54	0.	0.	2.91E+07	0.	0.	0.	0.	0.	0.	0.	0.
FE-55	0.	0.	5.55E+01	0.	0.	0.	0.	0.	0.	0.	0.
CO-60	0.	0.	1.22E+05	0.	0.	0.	0.	0.	0.	0.	0.
NI-59	0.	0.	1.08E+04	1.46E+04	1.25E+04	1.40E+04	1.34E+04	9.20E+03	4.18E+03	1.93E+02	2.54E+00
NI-63	0.	0.	9.75E+06	4.59E+05	1.26E+04	4.85E+10	0.	0.	0.	0.	0.
SE-79	0.	0.	1.13E+05	1.12E+05	1.11E+05	1.27E+05	1.01E+05	6.21E+04	3.88E+04	5.47E+02	2.49E+00
KR-85	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
RB-87	0.	0.	1.64E+01	1.64E+01	1.64E+01	1.64E+01	1.64E+01	1.64E+01	1.64E+01	1.64E+01	1.64E+01
SR=00+Y=90	0.	0.	4.42E+10	2.17E+06	9.43E+00	0.	0.	0.	0.	0.	0.
ZR=03	0.	0.	2.12E+03	2.12E+03	2.12E+03	2.11E+03	2.11E+03	2.07E+03	2.02E+03	1.64E+03	1.33E+03
NB=03M	0.	0.	4.04E+03	4.23E+03	4.24E+03	4.23E+03	4.22E+03	4.14E+03	4.05E+03	3.36E+03	2.67E+03
TC=09	0.	0.	4.19E+04	4.18E+04	4.18E+04	4.12E+04	4.06E+04	3.99E+04	3.01E+04	8.06E+03	1.55E+03
HU=106+RH=106	0.	0.	2.55E+01	0.	0.	0.	0.	0.	0.	0.	0.
PD=107	0.	0.	3.03E+04	3.02E+04	3.02E+04	3.02E+04	3.02E+04	3.01E+04	3.00E+04	3.26E+04	3.10E+04
AG=110M	0.	0.	2.79E+08	0.	0.	0.	0.	0.	0.	0.	0.
CO=113M	0.	0.	2.72E+06	5.14E+03	0.	0.	0.	0.	0.	0.	0.
SB=125+TE=125M	0.	0.	8.66E+02	0.	0.	0.	0.	0.	0.	0.	0.
SN=126+SB=126	0.	0.	4.47E+05	4.45E+05	4.44E+05	4.70E+05	4.44E+05	3.44E+05	2.44E+05	1.53E+04	4.78E+02
I=129	0.	0.	5.53E+05	5.53E+05	5.53E+05	5.53E+05	4.43E+05	4.22E+05	5.41E+05	9.42E+05	9.31E+05
CS=130	0.	0.	1.05E+06	0.	0.	0.	0.	0.	0.	0.	0.
CS=135	0.	0.	2.40E+03	2.40E+03	2.40E+03	2.40E+03	2.40E+03	2.77E+03	2.74E+03	2.90E+03	2.23E+03
CS=137+RA=137	0.	0.	1.12E+09	9.51E+04	9.21E+01	0.	0.	0.	0.	0.	0.
CE=144+PR=144	0.	0.	8.70E+04	0.	0.	0.	0.	0.	0.	0.	0.
PH=147	0.	0.	4.09E+03	0.	0.	0.	0.	0.	0.	0.	0.
SM=151	0.	0.	1.77E+06	6.99E+04	1.31E+03	1.22E+11	0.	0.	0.	0.	0.
EU=152	0.	0.	8.04E+03	5.33E+07	0.	0.	0.	0.	0.	0.	0.
EU=154	0.	0.	2.04E+07	9.78E+01	2.29E+10	0.	0.	0.	0.	0.	0.
EU=155	0.	0.	4.14E+01	0.	0.	0.	0.	0.	0.	0.	0.
OTHER	0.	0.	1.28E+10	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	0.	0.	4.03E+10	4.04E+06	1.26E+06	1.23E+04	1.21E+06	1.25E+04	9.13E+05	6.07E+05	5.71E+05
URANIUM ORE INOEX	0.	0.	5.67E+02	4.65E+02	1.45E+02	1.41E+02	1.40E+02	1.21E+02	1.05E+02	6.97E+03	6.56E+03

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.

TABLE A.4.7b. Hazard Index--Reprocessing Cycle--Growth Case 3--2010 Reprocessing Startup, m³ water/MTHM(A)

RADIONUCLIDES (A)	YEAR			REPLUGIC TIME (YEAR BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
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CM-245	0.	0.	1.04E+05	1.91E+05	1.23E+05	1.31E+05	3.42E+04	3.01E+03	4.54E+01	0.	0.
CM-244	0.	0.	7.44E+07	1.36E+01	6.44E+03	0.	0.	0.	0.	0.	0.
CM-243	0.	0.	3.15E+05	4.77E+01	2.03E+04	0.	0.	0.	0.	0.	0.
CM-242	0.	0.	7.74E+05	1.21E+05	1.24E+04	1.49E+04	0.	0.	0.	0.	0.
AM-243+MP-239	0.	0.	6.08E+06	5.46E+06	5.40E+06	3.90E+04	3.04E+06	4.10E+04	7.11E+02	0.	0.
AM-242+AM-242	0.	0.	4.01E+06	7.70E+05	7.88E+04	9.01E+04	0.	0.	0.	0.	0.
AM-241	0.	0.	6.55E+08	3.47E+08	1.44E+04	3.95E+09	4.49E+04	3.01E+03	4.55E+01	0.	0.
PU-242	0.	0.	4.33E+03	4.53E+03	4.49E+03	4.46E+03	4.42E+03	4.30E+03	3.44E+03	1.45E+03	7.40E+02
PU-241	0.	0.	5.06E+05	3.43E+03	3.47E+03	2.45E+03	1.73E+03	4.13E+01	9.10E+01	0.	0.
PU-240	0.	0.	1.91E+06	2.01E+06	1.01E+06	1.27E+04	7.49E+05	1.36E+04	7.45E+01	0.	0.
PU-239	0.	0.	6.22E+05	6.62E+05	7.19E+05	1.02E+04	1.21E+04	4.04E+05	1.41E+05	1.97E+00	1.27E+06
PU-238	0.	0.	5.05E+06	9.02E+05	1.11E+05	1.04E+03	0.	0.	0.	0.	0.
PU-236	0.	0.	3.49E+02	0.	0.	0.	0.	0.	0.	0.	0.
NP-237+PA-233	0.	0.	1.02E+05	2.21E+03	2.01E+05	3.26E+05	3.24E+05	3.23E+05	3.17E+05	2.79E+05	2.37E+05
U-238+Th-234+Pa-234m	0.	0.	1.02E+02	1.92E+02	1.02E+02	1.02E+02	1.02E+02	1.02E+02	1.02E+02	1.92E+02	1.92E+02
U-236	0.	0.	6.49E+01	7.04E+01	7.45E+01	1.35E+02	1.30E+02	1.45E+02	1.45E+02	1.63E+02	1.41E+02
U-235+Th-231	0.	0.	5.34E+00	5.43E+00	5.40E+00	5.14E+00	7.23E+00	1.08E+01	1.41E+01	1.92E+01	1.91E+01
U-234	0.	0.	4.04E+02	8.38E+02	1.00E+03	1.31E+03	0.09E+02	0.11E+02	7.05E+02	3.16E+02	1.02E+02
U-233	0.	0.	2.23E+00	3.00E+01	4.47E+01	5.23E+02	1.27E+03	4.10E+03	1.04E+04	2.51E+04	2.44E+04
U-232	0.	0.	4.11E+01	4.28E+01	4.49E+03	0.	0.	0.	0.	0.	0.
PA-231	0.	0.	3.42E+01	3.91E+01	4.45E+01	5.02E+01	4.44E+01	3.36E+02	4.02E+02	5.30E+02	5.53E+02
Th-230	0.	0.	1.11E+02	1.42E+02	2.08E+02	7.16E+02	1.33E+03	4.10E+03	7.76E+03	4.12E+03	2.44E+03
Th-229+ DAUGHTERS	0.	0.	1.28E+01	1.56E+01	9.04E+01	3.43E+03	1.31E+04	1.45E+05	2.94E+05	7.58E+05	7.36E+05
Th-228+ DAUGHTERS	0.	0.	8.92E+01	1.77E+00	1.00E+02	3.44E+00	4.70E+00	5.04E+03	1.15E+02	5.91E+02	1.18E+01
Ac-227+ DAUGHTERS	0.	0.	4.92E+01	6.95E+01	7.34E+01	9.04E+01	1.17E+02	4.04E+02	7.18E+02	9.90E+02	9.49E+02
Th-232+ DAUGHTERS	0.	0.	1.92E+04	1.34E+03	3.20E+03	1.49E+02	4.24E+02	2.47E+01	4.40E+01	2.48E+00	5.75E+00
Pa-226+ DAUGHTERS	0.	0.	1.74E+02	1.37E+03	3.40E+03	2.99E+04	7.04E+04	3.04E+05	4.27E+05	4.12E+05	1.71E+05
Pb-210+ DAUGHTERS	0.	0.	3.23E+01	4.43E+02	1.19E+03	1.02E+04	2.01E+04	1.17E+05	1.79E+05	1.40E+05	3.42E+04
TOTAL	0.	0.	7.40E+06	3.58E+05	1.45E+08	7.09E+04	4.06E+06	1.44E+06	1.41E+06	1.62E+06	1.23E+06
URANIUM ORE INDEX	0.	0.	8.74E+00	4.11E+00	1.49E+00	4.15E+02	4.42E+02	1.03E+02	1.73E+02	1.87E+02	1.42E+02

A. VALUES LESS THAN 1.0E+10 HAVE BEEN DESIGNATED AS ZERO.
 B. Th-229, 7 DAUGHTERS ARE PA-225, AC-224, FR-221, AT-217, AT-213, PB-204 AND TL-209 IS 9% OF Th-229 AND PO-213 IS 91% OF Th-229.
 Th-228, 6 DAUGHTERS ARE PA-224, RN-220, PO-216, PS-212, PT-212 AND TL-209 IS 34% OF Th-228 AND PO-212 IS 66% OF Th-228.
 AC-227, 7 DAUGHTERS ARE Th-227, HA-223, RN-219, PO-215, PR-211, PI-211 AND TL-207.
 Th-232, 2 DAUGHTERS ARE PA-228 AND AC-228.
 Pa-226, 5 DAUGHTERS ARE RN-222, PO-214, PD-214, BI-214 AND Pb-210.
 Pb-210, 2 DAUGHTERS ARE BI-210 AND PO-210.

NOTE. IN ACCOUNTING FOR THE ACTIVITY IN THIS MANNER, BRANCHING DECAY TO THE CASE OF TL-208 (34%) = PU-212 (44%), AND TL-209 (9%) = PO-211 (61%) WERE COUNTED AS A SINGLE DAUGHTER IN EACH CASE. MINOR BRANCHING (1% OR LESS) WAS IGNORED.

TABLE A.4.8a. Hazard Index--Reprocessing Cycle--Growth Case 4--2000 Reprocessing Startup, m³ water/MTHM(A)

Fission and Activation Products

MAJOR RADIONUCLIDES	YEAR			REPOSIC TIME (YEARS BEYOND 1975)								
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000	
H-3	0.	0.	1.21E+73	1.51E+07	0.	0.	0.	0.	0.	0.	0.	0.
C-14	0.	0.	1.04E+73	9.80E+02	9.23E+02	5.69E+02	3.11E+02	2.07E+00	5.26E+03	0.	0.	
HN-54	0.	0.	1.81E+76	0.	0.	0.	0.	0.	0.	0.	0.	0.
FE-55	0.	0.	1.84E+02	0.	0.	0.	0.	0.	0.	0.	0.	0.
CO-60	0.	0.	2.73E+05	0.	0.	0.	0.	0.	0.	0.	0.	0.
NI-59	0.	0.	1.37E+74	1.36E+04	1.36E+04	1.31E+04	1.26E+04	2.24E+03	5.75E+03	1.80E+02	2.38E+00	
NI-63	0.	0.	9.95E+06	4.07E+05	1.04E+04	5.62E+10	0.	0.	0.	0.	0.	0.
SE-79	0.	0.	1.10E+75	1.09E+05	1.09E+05	1.04E+05	9.88E+04	1.245E+04	3.78E+04	5.33E+02	2.99E+00	
KR-85	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
RB-87	0.	0.	1.55E+01	1.55E+01	1.55E+01	1.55E+01	1.55E+01	1.55E+01	1.55E+01	1.55E+01	1.55E+01	1.55E+01
SR-90+Y-90	0.	0.	5.28E+10	2.38E+06	1.25E+01	0.	0.	0.	0.	0.	0.	0.
ZR-93	0.	0.	2.05E+03	2.05E+03	2.05E+03	2.04E+03	2.04E+03	2.00E+03	1.96E+03	1.63E+03	1.29E+03	
NR-93m	0.	0.	3.84E+03	4.10E+03	4.10E+03	4.09E+03	4.08E+03	4.01E+03	3.92E+03	3.24E+03	2.59E+03	
TC-99	0.	0.	4.18E+04	4.18E+04	4.15E+04	4.10E+04	4.03E+04	3.93E+04	2.89E+04	1.00E+03	1.54E+03	
RU-106+RH-106	0.	0.	1.72E+00	0.	0.	0.	0.	0.	0.	0.	0.	0.
PO-107	0.	0.	3.78E+04	3.78E+04	3.76E+04	3.74E+04	3.76E+04	3.74E+04	3.73E+04	3.54E+04	3.41E+04	
AG-110M	0.	0.	2.77E+07	0.	0.	0.	0.	0.	0.	0.	0.	0.
CD-113M	0.	0.	4.07E+06	4.31E+03	0.	0.	0.	0.	0.	0.	0.	0.
BR-125+TE-125M	0.	0.	3.30E+03	0.	0.	0.	0.	0.	0.	0.	0.	0.
SN-126+SB-126	0.	0.	5.19E+05	5.18E+05	5.16E+05	5.02E+05	4.85E+05	3.24E+05	2.80E+05	1.63E+04	5.11E+02	
I-129	0.	0.	5.66E+05	5.66E+05	5.66E+05	5.66E+05	5.66E+05	5.66E+05	5.66E+05	5.54E+05	5.43E+05	
CS-134	0.	0.	4.45E+06	0.	0.	0.	0.	0.	0.	0.	0.	0.
CS-135	0.	0.	3.08E+03	3.08E+03	3.08E+03	3.08E+03	3.08E+03	3.05E+03	3.01E+03	2.75E+03	2.45E+03	
CS-137+CA-137	0.	0.	1.29E+09	1.09E+05	1.06E+00	0.	0.	0.	0.	0.	0.	0.
CE-140+PR-144	0.	0.	5.22E+03	0.	0.	0.	0.	0.	0.	0.	0.	0.
PM-147	0.	0.	1.35E+04	0.	0.	0.	0.	0.	0.	0.	0.	0.
SM-151	0.	0.	1.91E+06	7.53E+04	1.01E+03	2.07E+11	0.	0.	0.	0.	0.	0.
EU-152	0.	0.	1.03E+04	9.51E+07	0.	0.	0.	0.	0.	0.	0.	0.
EU-154	0.	0.	3.47E+07	8.30E+01	3.20E+10	0.	0.	0.	0.	0.	0.	0.
EU-155	0.	0.	2.12E+02	0.	0.	0.	0.	0.	0.	0.	0.	0.
DTMFR	0.	0.	7.18E+10	0.	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	0.	0.	5.01E+10	4.30E+06	1.31E+06	1.27E+04	1.25E+04	1.20E+06	9.03E+05	6.23E+05	5.86E+05	
URANIUM ORE INDEX	0.	0.	6.22E+02	4.95E+02	1.50E+02	1.45E+02	1.44E+02	1.35E+02	1.04E+02	7.14E+03	6.73E+03	

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.

TABLE A.4.8b. Hazard Index--Reprocessing Cycle--Growth Case 4--2000 Reprocessing Startup, m³ water/MTHM(A)

Actinides

RADIOISOTOPES (A)	YEAR			REQUIRE TIME YEARS BEYOND 1975							
	2000	2050	2170	500	1000	5000	10000	50000	100000	500000	1000000
CM-245	0.	0.	4.77E+05	4.63E+05	4.62E+05	4.16E+05	3.68E+05	7.05E+03	1.09E+02	0.	0.
CM-244	0.	0.	1.04E+06	3.43E+01	1.44E+07	0.	0.	0.	0.	0.	0.
CM-243	0.	0.	5.14E+05	7.79E+01	1.44E+03	0.	0.	0.	0.	0.	0.
CM-242	0.	0.	1.40E+06	2.50E+05	2.58E+04	3.07E+04	0.	0.	0.	0.	0.
AM-243+PM-239	0.	0.	1.43E+07	1.04E+07	9.05E+06	6.93E+06	4.00E+06	1.17E+05	1.26E+03	0.	0.
AM-242+AM-242	0.	0.	1.71E+07	1.54E+06	1.62E+05	1.94E+03	0.	0.	0.	0.	0.
AM-241	0.	0.	4.70E+08	2.46E+05	1.11E+08	5.13E+05	3.08E+05	7.08E+03	1.10E+02	0.	0.
Pl-242	0.	0.	7.19E+03	7.40E+03	7.92E+03	7.44E+03	7.41E+03	1.07E+03	4.05E+03	3.10E+03	1.24E+03
PU-241	0.	0.	1.10E+06	9.22E+03	1.25E+03	6.33E+03	4.14E+03	1.05E+02	2.19E+00	0.	0.
PU-240	0.	0.	3.45E+06	4.48E+05	4.17E+06	2.77E+04	1.44E+05	2.70E+04	1.43E+02	0.	0.
PU-239	0.	0.	7.47E+05	8.02E+04	9.07E+05	1.53E+06	1.09E+06	1.08E+06	2.47E+05	3.10E+00	2.10E+06
Pl-238	0.	0.	1.21E+07	2.11E+06	2.01E+05	2.96E+03	0.	0.	0.	0.	0.
Pl-236	0.	0.	2.22E+01	0.	0.	0.	0.	0.	0.	0.	0.
NP-237+PA-233	0.	0.	1.00E+05	2.47E+05	2.09E+05	3.21E+05	7.02E+05	7.10E+05	3.10E+05	2.74E+05	2.34E+05
U-238+Th-234+Pa-234m	0.	0.	1.05E+02	1.95E+02	1.05E+02	1.05E+02	1.05E+02	1.05E+02	1.05E+02	1.94E+02	1.95E+02
U-236	0.	0.	8.60E+01	9.37E+01	1.00E+02	1.70E+02	3.03E+02	7.00E+02	3.01E+02	2.97E+02	2.93E+02
U-235+Th-231	0.	0.	5.24E+00	5.39E+00	5.04E+00	6.42E+00	4.04E+00	2.04E+00	2.54E+00	2.75E+00	2.75E+00
U-234	0.	0.	5.74E+02	1.67E+03	1.02E+04	1.03E+04	1.01E+04	1.02E+04	1.40E+03	5.47E+02	2.00E+02
U-233	0.	0.	3.21E+00	3.08E+01	9.46E+01	6.09E+02	1.04E+03	5.04E+03	1.07E+04	2.40E+04	2.01E+04
U-232	0.	0.	7.23E+01	1.45E+00	1.17E+02	0.	0.	0.	0.	0.	0.
PA-231	0.	0.	3.42E+01	3.01E+01	4.04E+01	5.10E+01	4.71E+01	2.02E+02	4.54E+02	7.97E+02	7.97E+02
Th-230	0.	0.	6.17E+01	1.18E+02	2.45E+02	1.22E+03	2.40E+03	2.43E+03	1.07E+04	1.10E+04	3.41E+03
Th-229+ DAUGHTERS	0.	0.	2.11E+01	2.01E+01	1.45E+02	3.53E+03	1.72E+04	1.03E+04	2.03E+05	7.40E+05	7.27E+04
Th-228+ DAUGHTERS	0.	0.	1.55E+02	3.11E+00	2.43E+02	5.04E+00	1.13E+03	7.03E+03	1.51E+02	7.81E+02	1.46E+01
AC-227+ DAUGHTERS	0.	0.	5.19E+01	6.95E+01	7.23E+01	9.12E+01	1.00E+02	5.17E+02	9.07E+02	1.40E+03	1.02E+03
Th-227+ DAUGHTERS	0.	0.	1.44E+04	1.75E+03	4.14E+03	2.05E+02	5.53E+02	3.42E+01	7.39E+01	3.81E+00	7.40E+00
RA-226+ DAUGHTERS	0.	0.	1.00E+02	9.40E+02	3.04E+03	4.80E+04	1.04E+05	4.47E+05	9.07E+05	7.39E+05	2.56E+05
PB-210+ DAUGHTERS	0.	0.	1.01E+01	3.04E+02	1.11E+03	1.45E+04	4.08E+04	2.31E+04	3.40E+05	2.52E+05	8.73E+04
TOTAL	0.	0.	7.05E+08	2.67E+06	1.27E+08	1.25E+07	4.80E+06	2.54E+06	2.29E+04	2.04E+06	1.34E+06
URANIUM ORE INDEX	0.	0.	8.11E+00	3.07E+00	1.06E+00	1.43E+01	3.02E+01	2.07E+02	2.54E+02	2.36E+02	1.54E+02

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.
B. Th-229, 7 DAUGHTERS ARE RA-225, AC-225, FM-221, AT-217, HT-213, PA-209 AND TL-206 IS 9% OF Th-229 AND PO-215 IS 41% OF Th-229.
Th-228, 6 DAUGHTERS ARE RA-224, RN-220, PO-216, PS-212, OT-212 AND TL-208 IS 34% OF Th-228 AND PO-212 IS 60% OF Th-228.
AC-227, 7 DAUGHTERS ARE Th-227, RA-223, RN-219, PO-215, PA-211, PI-211 AND TL-207.
Th-222, 2 DAUGHTERS ARE RA-228 AND AC-228.
PA-226, 5 DAUGHTERS ARE RN-222, RN-218, PR-214, RI-214 AND PO-210.
PB-210, 2 DAUGHTERS ARE HI-210 AND PO-210.

NOTE. IN ACCOUNTING FOR THE ACTIVITY IN THIS MANNER, BRANCHING DECAY IN THE CASE OF TL-208 (34%) = PO-212 (44%), AND TL-204 (9%) = PO-211 (91%) WERE COUNTED AS A SINGLE DAUGHTER IN EACH CASE. OTHER BRANCHING (BY OR LESS) WAS IGNORED.

TABLE A.4.9a. Hazard Index--Reprocessing Cycle--Growth Case 5--2000 Reprocessing Startup, m³ water/MTHM(A)

Fission and Activation Products

MAJOR RADIOISOTOPES	YEAR			GEOLOGIC TIME (YEARS BEYOND 1975)								
	2000	2050	2170	500	1000	5000	10000	50000	100000	500000	1000000	
H-3	0.	0.	1.43E+03	1.40E+07	0.	0.	0.	0.	0.	0.	0.	0.
C-14	0.	0.	1.03E+03	9.81E+02	9.53E+02	5.49E+02	3.11E+02	2.07E+02	1.46E+02	0.	0.	
MN-54	0.	0.	2.52E+06	0.	0.	0.	0.	0.	0.	0.	0.	0.
FE-55	0.	0.	2.54E+02	0.	0.	0.	0.	0.	0.	0.	0.	0.
CO-60	0.	0.	3.43E+05	0.	0.	0.	0.	0.	0.	0.	0.	0.
NI-59	0.	0.	1.27E+04	1.45E+04	1.36E+04	1.31E+04	1.25E+04	1.20E+04	1.15E+04	1.10E+04	1.05E+04	1.00E+04
NI-63	0.	0.	9.78E+05	4.59E+05	1.06E+04	8.45E+10	0.	0.	0.	0.	0.	0.
SF-79	0.	0.	1.10E+05	1.09E+05	1.09E+05	1.08E+05	1.07E+05	1.06E+05	1.05E+05	1.04E+05	1.03E+05	1.02E+05
KR-85	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
Rb-87	0.	0.	1.54E+01	1.54E+01	1.54E+01	1.54E+01	1.54E+01	1.54E+01	1.54E+01	1.54E+01	1.54E+01	1.54E+01
SR-90+Y-90	0.	0.	5.73E+10	2.48E+06	1.14E+01	0.	0.	0.	0.	0.	0.	0.
ZR-93	0.	0.	2.07E+03	2.05E+03	2.05E+03	2.05E+03	2.04E+03	2.00E+03	1.96E+03	1.63E+03	1.29E+03	1.02E+03
NB-93M	0.	0.	3.74E+03	4.10E+03	4.10E+03	4.09E+03	4.08E+03	4.07E+03	4.02E+03	3.26E+03	2.59E+03	2.09E+03
TC-99	0.	0.	4.17E+04	4.18E+04	4.17E+04	4.12E+04	4.05E+04	3.95E+04	3.81E+04	3.04E+04	1.55E+04	1.05E+04
RU-106+RH-106	0.	0.	2.03E+00	0.	0.	0.	0.	0.	0.	0.	0.	0.
PO-107	0.	0.	3.84E+04	3.88E+04	3.88E+04	3.85E+04	3.80E+04	3.80E+04	3.82E+04	3.67E+04	3.49E+04	3.49E+04
AG-110M	0.	0.	4.01E+07	0.	0.	0.	0.	0.	0.	0.	0.	0.
CO-113M	0.	0.	5.24E+06	9.88E+03	0.	0.	0.	0.	0.	0.	0.	0.
SB-125+Y-125M	0.	0.	4.54E+03	0.	0.	0.	0.	0.	0.	0.	0.	0.
BN-126+BR-126	0.	0.	5.27E+05	5.20E+05	5.20E+05	5.11E+05	4.94E+05	4.79E+05	4.65E+05	1.66E+04	5.20E+02	5.20E+02
I-129	0.	0.	5.72E+05	5.72E+05	5.72E+05	5.72E+05	5.72E+05	5.71E+05	5.70E+05	5.60E+05	5.49E+05	5.49E+05
CS-134	0.	0.	6.75E+06	0.	0.	0.	0.	0.	0.	0.	0.	0.
CS-135	0.	0.	3.14E+03	3.14E+03	3.14E+03	3.14E+03	3.13E+03	3.10E+03	3.07E+03	2.80E+03	2.49E+03	2.49E+03
CS-137+BA-137	0.	0.	1.01E+09	1.14E+05	1.14E+00	0.	0.	0.	0.	0.	0.	0.
CE-140+BR-140	0.	0.	7.18E+03	0.	0.	0.	0.	0.	0.	0.	0.	0.
PH-147	0.	0.	1.81E+04	0.	0.	0.	0.	0.	0.	0.	0.	0.
SM-151	0.	0.	1.24E+06	7.02E+04	1.08E+03	2.15E+11	0.	0.	0.	0.	0.	0.
EU-152	0.	0.	1.72E+04	1.14E+06	0.	0.	0.	0.	0.	0.	0.	0.
EU-154	0.	0.	4.10E+07	9.87E+01	3.04E+10	0.	0.	0.	0.	0.	0.	0.
EU-155	0.	0.	2.97E+02	0.	0.	0.	0.	0.	0.	0.	0.	0.
DTMFR	0.	0.	1.05E+09	0.	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	0.	0.	5.07E+10	4.55E+06	1.22E+06	1.29E+04	1.27E+06	1.10E+06	9.55E+05	4.30E+05	5.92E+05	5.92E+05
URANIUM OPE INDEX	0.	0.	8.75E+02	5.23E+02	1.42E+02	1.88E+02	1.45E+02	1.27E+02	1.10E+02	7.24E+03	6.81E+03	6.81E+03

A. VALUES LESS THAN 1.0E-10 HAVE BEEN DESIGNATED AS ZERO.

TABLE A.4.9b. Hazard Index--Reprocessing Cycle--Growth Case 5--2000 Reprocessing Startup, m³ water/MTHM(A)

Actinides

RADIONUCLIDES (N)	YEAR			REPOSITORY LIFE (YEARS BEYOND 1975)							
	2000	2050	2070	500	1000	5000	10000	50000	100000	500000	1000000
CM-245	0.	0.	5.54E+05	5.40E+05	5.18E+05	3.71E+05	2.80E+05	2.21E+05	1.24E+02	0.	0.
CM-244	0.	0.	2.87E+07	4.36E+01	2.11E+07	0.	0.	0.	0.	0.	0.
CM-243	0.	0.	5.95E+05	2.01E+01	1.78E+05	0.	0.	0.	0.	0.	0.
CM-242	0.	0.	1.84E+05	2.95E+00	3.02E+04	3.41E+04	0.	0.	0.	0.	0.
AM-243+MP-239	0.	0.	1.29E+07	1.20E+07	1.14E+07	7.09E+04	5.05E+04	1.35E+05	1.04E+03	0.	0.
AM-242M+AM-242	0.	0.	1.19E+07	1.87E+06	1.01E+05	2.04E+04	0.	0.	0.	0.	0.
AM-241	0.	0.	4.74E+09	2.42E+04	1.13E+03	5.75E+05	2.04E+05	8.52E+03	1.29E+02	0.	0.
PU-242	0.	0.	1.37E+04	1.13E+00	1.10E+00	1.14E+00	1.13E+04	1.09E+00	9.55E+03	4.00E+03	1.04E+03
PU-241	0.	0.	1.89E+06	1.36E+04	1.04E+04	7.02E+03	0.49E+03	1.17E+02	2.57E+00	0.	0.
PU-240	0.	0.	4.50E+06	5.19E+06	4.05E+06	3.28E+04	1.07E+04	3.35E+04	1.93E+02	0.	0.
PU-239	0.	0.	8.04E+05	9.36E+05	1.06E+05	1.72E+04	2.19E+04	3.31E+04	3.04E+05	3.53E+00	2.39E+04
PU-238	0.	0.	1.07E+07	2.07E+05	2.04E+05	3.09E+03	0.	0.	0.	0.	0.
PU-236	0.	0.	4.34E+02	0.	0.	0.	0.	0.	0.	0.	0.
NP-237+PA-233	0.	0.	1.74E+05	2.32E+05	2.75E+05	3.75E+05	3.09E+05	3.28E+05	3.02E+05	2.54E+05	2.25E+05
U-238+Th-234+Pa-234m	0.	0.	1.94E+02	1.96E+02	1.96E+02	1.96E+02	1.96E+02	1.96E+02	1.96E+02	1.96E+02	1.97E+02
U-236	0.	0.	7.02E+01	4.92E+01	1.01E+02	1.79E+02	2.02E+02	3.19E+02	3.74E+02	3.30E+02	3.25E+02
U-235+Th-231	0.	0.	5.14E+00	5.25E+00	5.45E+00	6.41E+00	6.27E+00	2.33E+01	2.45E+01	3.75E+01	3.03E+01
U-234	0.	0.	5.69E+02	1.72E+03	2.13E+03	2.16E+03	2.14E+03	1.01E+03	1.47E+03	4.04E+02	2.14E+02
U-233	0.	0.	2.02E+00	3.40E+01	4.94E+01	5.02E+02	1.01E+03	5.21E+03	1.03E+04	2.39E+04	2.41E+04
U-232	0.	0.	1.61E+04	3.22E+02	2.62E+00	0.	0.	0.	0.	0.	0.
PA-231	0.	0.	3.43E+01	3.62E+01	3.75E+01	4.81E+01	4.08E+01	3.09E+02	6.11E+02	4.92E+02	4.02E+02
Th-230	0.	0.	5.54E+01	1.16E+02	2.07E+02	1.75E+04	2.46E+03	1.07E+04	1.43E+04	1.21E+04	4.12E+03
Th-229+7 DAUGHTERS	0.	0.	1.74E+01	1.83E+01	1.07E+02	3.36E+03	1.04E+04	1.24E+05	2.82E+05	7.20E+05	6.09E+05
Th-228+6 DAUGHTERS	0.	0.	3.03E+04	6.91E+02	5.61E+00	4.90E+00	1.12E+03	7.07E+03	1.55E+02	4.18E+02	1.43E+01
Ac-227+7 DAUGHTERS	0.	0.	4.47E+01	6.43E+01	6.70E+01	6.60E+01	1.16E+02	5.53E+02	1.09E+03	1.58E+03	1.58E+03
Th-232+2 DAUGHTERS	0.	0.	1.04E+04	1.61E+03	3.04E+03	2.34E+02	5.05E+02	1.20E+01	7.49E+01	3.99E+00	7.96E+00
Pa-226+5 DAUGHTERS	0.	0.	8.44E+01	4.92E+02	3.09E+03	5.30E+04	1.39E+05	7.04E+05	1.11E+05	4.17E+05	2.77E+05
Pb-210+2 DAUGHTERS	0.	0.	1.42E+01	2.84E+02	1.12E+03	1.40E+04	4.74E+04	2.08E+05	3.78E+05	2.78E+05	9.04E+04
TOTAL	0.	0.	7.75E+04	2.76E+04	1.82E+03	1.43E+07	1.02E+07	2.04E+05	2.02E+04	2.12E+06	1.33E+06
URANIUM ORE INDEX	0.	0.	6.91E+00	3.17E+00	1.02E+00	1.64E+01	1.17E+01	3.27E+02	2.75E+02	2.44E+02	1.53E+02

A. VALUES LESS THAN 1.0E+10 HAVE BEEN DESIGNATED AS ZERO.
B. Th-229, 7 DAUGHTERS ARE PA-225, AC-225, FR-221, AT-217, CT-213, PR-209 AND TL-200 IS 9% OF Th-229 AND 60-213 IS 91% OF Th-229.
Th-228, 6 DAUGHTERS ARE PA-224, FR-220, PU-216, PB-212, BI-212 AND TL-204 IS 14% OF Th-228 AND 60-213 IS 50% OF Th-228.
AC-227, 7 DAUGHTERS ARE Th-227, PA-223, NN-219, PD-215, CP-211, CT-211 AND TL-207.
Th-232, 2 DAUGHTERS ARE PA-228 AND AC-224.
Pa-226, 5 DAUGHTERS ARE NN-222, PP-214, PD-210, BI-210 AND PR-210.
Pb-210, 2 DAUGHTERS ARE AT-210 AND PD-210.
NOTE. IN ACCOUNTING FOR THE ACTIVITY IN THIS MANNER, BRANCHING DECAY IN THE CASE OF TL-209 (34%) - PB-212 (64%), AND TL-209 (6%) - PP-211 (61%) WERE COUNTED AS A SINGLE DAUGHTER IN EACH CASE. MINOR BRANCHING (1% OR LESS) WAS IGNORED.

A.5 SUPPLEMENTARY DOSE TABLES

The radiation dose tables (A.5.1a through A.5.2d) provide detail on regional population and world-wide doses. Each table, one for the once-through cycle and one for the reprocessing cycle, is composed of four tables. Each sub-table provides the whole-body, bone, lung and thyroid doses.

TABLE A.5.1a. Whole-Body Dose to the Population for the Once-Through Cycle, Man-Rem

Case	Growth Assumption	Repository Start-Up Date	Storage of Spent Fuel	BWR Fuel Shipments	PWR Fuel Shipments	Total
1	Present Inventory	None	2.18×10^{-1}	0	0	2.18×10^{-1}
1		1990	1.30×10^{-1}	1.64×10^1	1.97×10^1	3.62×10^1
1		2010	1.53×10^{-1}	1.64×10^1	1.97×10^1	3.62×10^1
1		2030	1.68×10^{-1}	1.64×10^1	1.97×10^1	3.63×10^1
2	Present Capacity to Retirement	None	1.63×10^0	3.32×10^1	5.57×10^1	9.05×10^1
2		1990	9.17×10^{-1}	8.21×10^1	1.18×10^2	2.01×10^2
2		2010	1.33×10^0	1.01×10^2	1.50×10^2	2.53×10^2
2		2030	1.54×10^0	1.05×10^2	1.57×10^2	2.64×10^2
3	250 GWe in 2000 and Decline to 0 in 2040	None	7.92×10^0	1.77×10^2	2.98×10^2	4.83×10^2
3		1990	4.41×10^0	3.65×10^2	5.66×10^2	9.35×10^2
3		2010	6.35×10^0	4.81×10^2	7.61×10^2	1.25×10^3
3		2030	7.61×10^0	5.10×10^2	8.10×10^2	1.33×10^3
4	250 GWe in 2000 and Steady to 2040	2000	6.73×10^0	5.49×10^2	8.64×10^2	1.42×10^3
4		2020	9.18×10^0	6.75×10^2	1.08×10^3	1.76×10^3
5	250 GWe in 2000 and 500 GWe in 2040	2000	8.68×10^0	7.20×10^2	1.14×10^3	1.87×10^3
5		2020	1.25×10^1	9.00×10^2	1.44×10^3	2.35×10^3

TABLE A.5.1b. Bone Dose to the Population for the Once-Through Cycle, Man-Rem

Case	Growth Assumption	Repository Start-Up Date	Storage of Spent Fuel	BWR Fuel Shipments	PWR Fuel Shipments	Total
1	Present Inventory	None	4.46×10^{-1}	0	0	4.46×10^{-1}
1		1990	2.55×10^{-1}	0	0	2.55×10^{-1}
1		2010	3.06×10^{-1}	0	0	3.06×10^{-1}
1		2030	3.41×10^{-1}	0	0	3.41×10^{-1}
2	Present Capacity to Retirement	None	2.30×10^0	0	0	2.30×10^0
2		1990	1.75×10^0	0	0	1.75×10^0
2		2010	2.63×10^0	0	0	2.63×10^0
2		2030	3.07×10^0	0	0	3.07×10^0
3	250 GWe in 2000 and Decline to 0 in 2040	None	1.58×10^1	0	0	1.58×10^1
3		1990	8.29×10^0	0	0	8.29×10^0
3		2010	1.24×10^1	0	0	1.24×10^1
3		2030	1.51×10^1	0	0	1.51×10^1
4	250 GWe in 2000 and Steady to 2040	2000	1.29×10^1	0	0	1.29×10^1
4		2020	1.81×10^1	0	0	1.81×10^1
5	250 GWe in 2000 and 500 GWe in 2040	2000	1.69×10^1	0	0	1.69×10^1
5		2020	2.46×10^1	0	0	2.46×10^1

TABLE A.5.1c. Lung Dose to the Population for the Once-Through Cycle, Man-Rem

Case	Growth Assumption	Repository Start-Up Date	Storage of Spent Fuel	BWR Fuel Shipments	PWR Fuel Shipments	Total
1	Present Inventory	None	9.85×10^{-2}	0	0	9.85×10^{-2}
1		1990	7.00×10^{-2}	0	0	7.00×10^{-2}
1		2010	7.22×10^{-2}	0	0	7.22×10^{-2}
1		2030	7.61×10^{-2}	0	0	7.61×10^{-2}
2	Present Capacity to Retirement	None	7.58×10^{-1}	0	0	7.58×10^{-1}
2		1990	5.14×10^{-1}	0	0	5.14×10^{-1}
2		2010	6.42×10^{-1}	0	0	6.42×10^{-1}
2		2030	7.18×10^{-1}	0	0	7.18×10^{-1}
3	250 GWe in 2000 and Decline to 0 in 2040	None	3.73×10^0	0	0	3.73×10^0
3		1990	2.58×10^0	0	0	2.58×10^0
3		2010	3.18×10^0	0	0	3.18×10^0
3		2030	3.59×10^0	0	0	3.59×10^0
4	250 GWe in 2000 and Steady to 2040	2000	3.70×10^0	0	0	3.70×10^0
4		2020	4.46×10^0	0	0	4.46×10^0
5	250 GWe in 2000 and 500 GWe in 2040	2000	4.92×10^0	0	0	4.92×10^0
5		2020	6.02×10^0	0	0	6.02×10^0

TABLE A.5.1d. Thyroid Dose to the Population for the Once-Through Cycle, Man-Rem

Case	Growth Assumption	Repository Start-Up Date	Storage of Spent Fuel	BWR Fuel Shipments	PWR Fuel Shipments	Total
1	Present Inventory	None	3.83×10^{-1}	0	0	3.83×10^{-1}
1		1990	3.93×10^0	0	0	3.93×10^0
1		2010	3.99×10^0	0	0	3.99×10^0
1		2030	4.04×10^0	0	0	4.04×10^0
2	Present Capacity to Retirement	None	2.38×10^0	0	0	2.38×10^0
2		1990	2.49×10^1	0	0	2.49×10^1
2		2010	2.55×10^1	0	0	2.55×10^1
2		2030	2.60×10^1	0	0	2.60×10^1
3	250 GWe in 2000 and Decline to 0 in 2040	None	1.06×10^1	0	0	1.06×10^1
3		1990	1.25×10^2	0	0	1.25×10^2
3		2010	1.27×10^2	0	0	1.27×10^2
3		2030	1.30×10^2	0	0	1.30×10^2
4	250 GWe in 2000 and Steady to 2040	2000	1.65×10^2	0	0	1.65×10^2
4		2020	1.69×10^2	0	0	1.69×10^2
5	250 GWe in 2000 and 500 GWe in 2040	2000	2.24×10^2	0	0	2.24×10^2
5		2020	2.29×10^2	0	0	2.29×10^2

TABLE A.5.2a. Whole-Body Dose to the Population for the Reprocessing Cycle, Man-Rem

Case	Growth Assumption	Reprocessing Start-up Date	Repository Start-Up Date	Storage of Spent Fuel	BWR Fuel Shipments	PWR Fuel Shipments	FRP Treatment System	MOX-FFP Treatment System	Reprocessing Waste Shipments	Total
3	250 GWe in 2000 and Decline to 0 in 2040	1990	1990	1.34×10^0	2.33×10^2	3.92×10^2	3.18×10^4	2.97×10^{-1}	5.97×10^2	3.31×10^4
3		1990	2010	1.34×10^0	2.33×10^2	3.92×10^2	3.18×10^4	2.97×10^{-1}	5.97×10^2	3.31×10^4
3		2010	2010	5.24×10^0	3.47×10^2	5.82×10^2	1.14×10^4	6.55×10^{-2}	5.78×10^2	1.29×10^4
3		1990	2030	1.34×10^0	2.33×10^2	3.92×10^2	3.18×10^4	2.97×10^{-1}	5.97×10^2	3.31×10^4
3		2010	2030	5.24×10^0	3.47×10^2	5.82×10^2	1.14×10^4	6.55×10^{-2}	5.78×10^2	1.29×10^4
4	250 GWe in 2000 and Steady State to 2040	2000	2000	3.84×10^0	3.78×10^2	6.35×10^2	3.07×10^4	2.91×10^{-1}	7.74×10^2	3.25×10^4
4		2000	2020	3.84×10^0	3.78×10^2	6.35×10^2	3.07×10^4	2.91×10^{-1}	7.74×10^2	3.25×10^4
5	250 GWe in 2000 and 500 GWe in 2040	2000	2000	4.73×10^0	4.94×10^2	8.30×10^2	4.34×10^4	5.03×10^{-1}	1.05×10^3	4.58×10^4
5		2000	2020	4.73×10^0	4.94×10^2	8.30×10^2	4.34×10^4	5.03×10^{-1}	1.05×10^3	4.58×10^4

TABLE A.5.2b. Bone Dose to the Population for the Reprocessing Cycle, Man-Rem

Case	Growth Assumption	Reprocessing Start-up Date	Repository Start-Up Date	Storage of Spent Fuel	BWR Fuel Shipments	PWR Fuel Shipments	FRP Treatment System	MOX-FFP Treatment System	Reprocessing Waste Shipments	Total
3	250 GWe in 2000 and Decline to 0 in 2040	1990	1990	2.01×10^0	0	0	2.50×10^4	6.43×10^0	0	2.50×10^4
3		1990	2010	2.01×10^0	0	0	2.50×10^4	6.43×10^0	0	2.50×10^4
3		2010	2010	9.15×10^0	0	0	8.55×10^2	1.42×10^0	0	8.66×10^2
3		1990	2030	2.01×10^0	0	0	2.50×10^4	6.43×10^0	0	2.50×10^4
3		2010	2030	9.15×10^0	0	0	8.55×10^2	1.42×10^0	0	8.66×10^2
4	250 GWe in 2000 and Steady State to 2040	2000	2000	6.31×10^0	0	0	1.49×10^4	6.31×10^0	0	1.49×10^4
4		2000	2020	6.31×10^0	0	0	1.49×10^4	6.31×10^0	0	1.49×10^4
5	250 GWe in 2000 and 500 GWe in 2040	2000	2000	7.67×10^0	0	0	1.81×10^4	1.09×10^1	0	1.81×10^4
5		2000	2020	7.67×10^0	0	0	1.81×10^4	1.09×10^1	0	1.81×10^4

TABLE A.5.2c. Lung Dose to the Population for the Reprocessing Cycle, Man-Rem

Case	Growth Assumption	Reprocessing Start-up Date	Repository Start-Up Date	Storage of Spent Fuel	BWR Fuel Shipments	PWR Fuel Shipments	FRP Treatment System	MOX-FFP Treatment System	Reprocessing Waste Shipments	Total
3	250 GWe in 2000 and Decline to 0 in 2040	1990	1990	6.44×10^{-1}	0	0	1.84×10^6	1.17×10^0	0	1.84×10^6
3		1990	2010	6.44×10^{-1}	0	0	1.84×10^6	1.17×10^0	0	1.84×10^6
3		2010	2010	2.53×10^0	0	0	4.34×10^4	2.56×10^{-1}	0	4.34×10^4
3		1990	2030	6.44×10^{-1}	0	0	1.84×10^6	1.17×10^0	0	1.84×10^6
3		2010	2030	2.53×10^0	0	0	4.34×10^4	2.56×10^{-1}	0	4.34×10^4
4	250 GWe in 2000 and Steady State to 2040	2000	2000	1.85×10^0	0	0	1.10×10^6	1.13×10^0	0	1.10×10^6
4		2000	2020	1.85×10^0	0	0	1.10×10^6	1.13×10^0	0	1.10×10^6
5	250 GWe in 2000 and 500 GWe in 2040	2000	2000	2.28×10^0	0	0	1.33×10^6	1.98×10^0	0	1.33×10^6
5		2000	2020	2.28×10^0	0	0	1.33×10^6	1.98×10^0	0	1.33×10^6

TABLE A.5.2d. Thyroid Dose to the Population for the Reprocessing Cycle, Man-Rem

Case	Growth Assumption	Reprocessing Start-up Date	Repository Start-Up Date	Storage of Spent Fuel	BWR Fuel Shipments	PWR Fuel Shipments	FRP Treatment System	MOX-FFP Treatment System	Reprocessing Waste Shipments	Total
3	250 GWe in 2000 and Decline to 0 in 2040	1990	1990	9.88×10^{-1}	0	0	4.82×10^4	3.49×10^{-9}	0	4.82×10^4
3		1990	2010	9.88×10^{-1}	0	0	4.82×10^4	3.49×10^{-9}	0	4.82×10^4
3		2010	2010	4.71×10^0	0	0	2.99×10^4	7.64×10^{-10}	0	2.99×10^4
3		1990	2030	9.88×10^{-1}	0	0	4.82×10^4	3.49×10^{-9}	0	4.82×10^4
3		2010	2030	4.71×10^0	0	0	2.99×10^4	7.64×10^{-10}	0	2.99×10^4
4	250 GWe in 2000 and Steady State to 2040	2000	2000	3.16×10^0	0	0	5.40×10^4	3.36×10^{-9}	0	5.40×10^4
4		2000	2020	3.16×10^0	0	0	5.40×10^4	3.36×10^{-9}	0	5.40×10^4
5	250 GWe in 2000 and 500 GWe in 2040	2000	2000	3.87×10^0	0	0	7.55×10^4	5.88×10^{-9}	0	7.55×10^4
5		2000	2020	3.87×10^0	0	0	7.55×10^4	5.88×10^{-9}	0	7.55×10^4

A.6 RESOURCE COMMITMENTS

Resource commitment tables (A.6.1 through A.6.3) list requirements by resource for all of the cases analyzed. The first table lists requirements for the once-through cycle; the second lists requirements for the reprocessing fuel cycles; and the third lists requirements for shipping casks.

TABLE A.6.1. Resource Commitments With the Once-Through Cycle

Growth Assumptions	Repository Startup Date	Repository Media	Steel, MT	Cement, MT	Diesel, M ³	Gasoline, M ³	Propane, M ³	Electricity, KW - hr	Manpower, Man - Year	
Present Inventory	None 1990	Salt	0	0	0	0	0	0	0	
		Granite	6.8×10^3	3.4×10^3	5.7×10^4	1.8×10^3	2.5×10^2	1.7×10^8	2.4×10^3	
		Basalt	1.4×10^4	4.4×10^3	5.3×10^4	2.5×10^3	3.4×10^2	1.9×10^8	2.8×10^3	
	2010	Salt	8.8×10^3	4.8×10^3	6.0×10^4	2.3×10^3	3.7×10^2	2.0×10^8	3.1×10^3	
		Granite	1.4×10^4	3.8×10^3	5.3×10^4	2.1×10^3	2.8×10^2	1.9×10^8	3.1×10^3	
		Basalt	6.8×10^3	3.4×10^3	5.7×10^4	1.8×10^3	2.5×10^2	1.7×10^8	2.4×10^3	
	2030	Salt	1.1×10^4	3.5×10^3	4.9×10^4	2.0×10^3	2.7×10^2	1.5×10^8	2.2×10^3	
		Granite	8.8×10^3	4.8×10^3	6.0×10^4	2.3×10^3	3.7×10^2	2.0×10^8	3.1×10^3	
		Basalt	1.1×10^4	3.0×10^3	4.9×10^4	1.7×10^3	2.2×10^2	1.5×10^8	2.5×10^3	
	Present Capacity to Retirement	None 1990	Salt	6.8×10^3	3.4×10^3	5.7×10^4	1.8×10^3	2.5×10^2	1.7×10^8	2.4×10^5
			Granite	8.4×10^3	2.6×10^3	4.5×10^4	1.5×10^3	2.0×10^2	1.1×10^8	1.7×10^3
			Basalt	8.8×10^3	4.8×10^3	6.0×10^4	2.3×10^3	3.7×10^2	2.0×10^8	3.1×10^3
2010		Salt	8.1×10^3	2.3×10^3	4.5×10^4	1.3×10^3	1.7×10^2	1.1×10^8	1.9×10^3	
		Granite	2.1×10^5	4.9×10^5	1.1×10^5	4.7×10^4	7.0×10^3	5.2×10^8	3.2×10^4	
		Basalt	9.2×10^4	1.3×10^5	3.4×10^5	2.3×10^4	3.3×10^3	1.3×10^9	2.4×10^4	
2030		Salt	1.3×10^5	1.3×10^5	2.9×10^5	2.5×10^4	3.5×10^3	1.2×10^9	2.3×10^4	
		Granite	8.8×10^4	1.2×10^5	3.1×10^5	2.1×10^4	3.4×10^3	1.1×10^9	2.3×10^4	
		Basalt	1.2×10^5	1.2×10^5	2.9×10^5	2.3×10^4	3.1×10^3	1.2×10^9	2.6×10^4	
250 GWe in 2000 and Decline to 0 in 2040		None 1990	Salt	2.2×10^5	4.2×10^5	4.1×10^5	5.1×10^4	7.5×10^3	1.6×10^9	4.3×10^4
			Granite	2.2×10^5	4.2×10^5	3.2×10^5	4.8×10^4	7.1×10^3	1.2×10^9	3.7×10^4
			Basalt	2.1×10^5	4.2×10^5	3.8×10^5	4.9×10^4	7.6×10^3	1.4×10^9	4.2×10^4
	2010	Salt	2.3×10^5	4.2×10^5	3.3×10^5	4.7×10^4	6.8×10^3	1.2×10^9	3.9×10^4	
		Granite	2.6×10^5	5.1×10^5	4.2×10^5	6.0×10^4	8.8×10^3	1.7×10^9	4.9×10^4	
		Basalt	2.6×10^5	5.1×10^5	3.3×10^5	5.7×10^4	8.4×10^3	1.3×10^9	4.3×10^4	
	2030	Salt	2.4×10^5	5.1×10^5	3.6×10^5	5.6×10^4	8.5×10^3	1.3×10^9	4.5×10^4	
		Granite	2.7×10^5	5.1×10^5	3.4×10^5	5.6×10^4	8.1×10^3	1.3×10^9	4.5×10^4	
		Basalt	1.1×10^6	2.6×10^6	6.0×10^5	2.5×10^5	3.7×10^4	2.8×10^9	1.7×10^5	
	250 GWe in 2000 and Steady State to 2040	None 1990	Salt	3.0×10^5	2.8×10^5	1.6×10^6	7.9×10^4	1.1×10^4	6.1×10^9	8.9×10^4
			Granite	4.9×10^5	3.0×10^5	1.4×10^6	8.6×10^4	1.3×10^4	5.8×10^9	9.4×10^4
			Basalt	2.9×10^5	2.9×10^5	1.5×10^6	7.5×10^4	1.2×10^4	5.4×10^9	8.6×10^4
2010		Salt	4.8×10^5	2.7×10^5	1.4×10^6	7.8×10^4	1.1×10^4	5.8×10^9	9.9×10^4	
		Granite	8.8×10^5	1.6×10^6	1.9×10^6	2.0×10^5	3.0×10^4	7.3×10^9	1.8×10^5	
		Basalt	1.0×10^6	1.6×10^6	1.6×10^6	2.1×10^5	2.9×10^4	6.4×10^9	1.7×10^5	
2030		Salt	8.6×10^5	1.6×10^6	1.8×10^6	2.0×10^5	3.0×10^4	6.4×10^9	1.7×10^5	
		Granite	1.0×10^6	1.6×10^6	1.5×10^6	1.9×10^5	2.8×10^4	6.4×10^9	1.8×10^5	
		Basalt	1.3×10^6	2.5×10^6	2.1×10^6	2.9×10^5	4.3×10^4	8.1×10^9	2.4×10^5	
250 GWe in 2000 and 500 GWe in 2040		2000	Salt	1.3×10^6	2.5×10^6	1.7×10^6	2.8×10^5	4.0×10^4	6.2×10^9	2.1×10^5
			Granite	1.2×10^6	2.5×10^6	1.9×10^6	2.8×10^5	4.2×10^4	6.7×10^9	2.2×10^5
			Basalt	1.2×10^6	2.5×10^6	1.6×10^6	2.7×10^5	3.9×10^4	6.2×10^9	2.2×10^5
	2020	Salt	6.6×10^5	9.3×10^5	2.3×10^6	1.6×10^5	2.4×10^4	9.3×10^9	1.7×10^5	
		Granite	9.0×10^5	9.4×10^5	1.9×10^6	1.7×10^5	2.4×10^4	7.4×10^9	1.6×10^5	
		Basalt	6.4×10^5	9.3×10^5	2.1×10^6	1.6×10^5	2.4×10^4	8.0×10^9	1.5×10^5	
	2030	Salt	8.1×10^5	9.5×10^5	1.9×10^6	1.6×10^5	2.3×10^4	7.4×10^9	1.7×10^5	
		Granite	1.4×10^6	2.4×10^6	2.7×10^6	3.0×10^5	4.5×10^4	1.0×10^{10}	2.7×10^5	
		Basalt	1.4×10^6	2.4×10^6	2.2×10^6	3.0×10^5	4.3×10^4	8.3×10^9	2.4×10^5	
	2000	Salt	1.2×10^6	2.4×10^6	2.4×10^6	2.9×10^5	4.4×10^4	8.2×10^9	2.4×10^5	
		Granite	1.4×10^6	2.4×10^6	2.2×10^6	2.9×10^5	4.2×10^4	8.3×10^9	2.5×10^5	
		Shale	7.8×10^5	1.0×10^6	3.0×10^6	1.9×10^5	2.8×10^4	1.2×10^{10}	2.1×10^5	
Basalt		1.1×10^6	1.0×10^6	2.6×10^6	1.8×10^5	2.7×10^4	1.1×10^{10}	2.1×10^5		
Shale		7.4×10^5	1.0×10^6	2.8×10^6	2.0×10^5	2.9×10^4	1.0×10^{10}	2.0×10^5		
Basalt		1.1×10^6	1.0×10^6	2.6×10^6	1.8×10^5	2.7×10^4	1.0×10^{10}	2.0×10^5		
2020	Salt	1.6×10^6	3.1×10^6	3.6×10^6	3.9×10^5	5.7×10^4	1.4×10^{10}	3.4×10^5		
	Granite	1.8×10^6	3.1×10^6	2.9×10^6	3.7×10^5	5.5×10^4	1.4×10^{10}	3.1×10^5		
	Shale	1.6×10^6	3.1×10^6	3.2×10^6	3.6×10^5	5.6×10^4	1.1×10^{10}	3.1×10^5		
	Basalt	1.7×10^6	3.1×10^6	2.9×10^6	3.6×10^5	5.3×10^4	1.1×10^{10}	3.2×10^5		
	Shale	1.6×10^6	3.1×10^6	3.2×10^6	3.6×10^5	5.6×10^4	1.1×10^{10}	3.1×10^5		
	Basalt	1.7×10^6	3.1×10^6	2.9×10^6	3.6×10^5	5.3×10^4	1.1×10^{10}	3.2×10^5		

TABLE A.6.2. Resource Commitments with the Reprocessing Cycle

Growth Assumptions	Reprocessing Startup Date	Repository Startup Date	Repository Media	Steel, MT	Cement, MT	Diesel, M ³	Gasoline, M ³	Propane, M ³	Electricity, KW - hr	Manpower, Man - Year
250 GWe in 2000 and Decline to 0 in 2040	1990	1990	Salt	4.8 x 10 ⁵	5.5 x 10 ⁵	1.4 x 10 ⁶	1.1 x 10 ⁵	3.5 x 10 ⁷	1.8 x 10 ¹⁰	1.4 x 10 ⁵
			Granite	7.2 x 10 ⁵	6.2 x 10 ⁵	1.4 x 10 ⁶	1.5 x 10 ⁵	3.5 x 10 ⁷	1.9 x 10 ¹⁰	1.8 x 10 ⁵
			Shale	3.8 x 10 ⁵	6.4 x 10 ⁵	1.6 x 10 ⁶	1.5 x 10 ⁵	3.5 x 10 ⁷	1.9 x 10 ¹⁰	1.8 x 10 ⁵
			Basalt	7.4 x 10 ⁵	6.1 x 10 ⁵	1.4 x 10 ⁶	1.5 x 10 ⁵	3.5 x 10 ⁷	1.8 x 10 ¹⁰	2.0 x 10 ⁵
	1990	2010	Salt	6.7 x 10 ⁵	5.9 x 10 ⁵	1.4 x 10 ⁶	6.3 x 10 ⁴	3.5 x 10 ⁷	1.8 x 10 ¹⁰	1.5 x 10 ⁵
			Granite	8.4 x 10 ⁵	6.4 x 10 ⁵	1.4 x 10 ⁶	2.3 x 10 ⁵	3.5 x 10 ⁷	1.8 x 10 ¹⁰	1.8 x 10 ⁵
			Shale	5.6 x 10 ⁵	6.7 x 10 ⁵	1.6 x 10 ⁶	2.4 x 10 ⁵	3.5 x 10 ⁷	1.8 x 10 ¹⁰	2.0 x 10 ⁵
			Basalt	8.6 x 10 ⁵	6.4 x 10 ⁵	1.4 x 10 ⁶	2.3 x 10 ⁵	3.5 x 10 ⁷	1.8 x 10 ¹⁰	1.9 x 10 ⁵
	2010	2010	Salt	9.3 x 10 ⁵	1.8 x 10 ⁶	1.3 x 10 ⁶	2.2 x 10 ⁵	3.4 x 10 ⁷	1.7 x 10 ¹⁰	2.0 x 10 ⁵
			Granite	1.1 x 10 ⁶	1.8 x 10 ⁶	1.3 x 10 ⁶	2.5 x 10 ⁵	3.4 x 10 ⁷	1.7 x 10 ¹⁰	2.3 x 10 ⁵
			Shale	9.0 x 10 ⁵	1.9 x 10 ⁶	1.5 x 10 ⁶	2.5 x 10 ⁵	3.4 x 10 ⁷	1.8 x 10 ¹⁰	2.4 x 10 ⁵
			Basalt	1.1 x 10 ⁶	1.8 x 10 ⁶	1.3 x 10 ⁶	2.5 x 10 ⁵	3.4 x 10 ⁷	1.7 x 10 ¹⁰	2.4 x 10 ⁵
1990	2030	Salt	7.2 x 10 ⁵	7.7 x 10 ⁵	1.7 x 10 ⁶	2.4 x 10 ⁵	3.5 x 10 ⁷	1.7 x 10 ¹⁰	1.5 x 10 ⁵	
		Granite	8.7 x 10 ⁵	8.1 x 10 ⁵	1.8 x 10 ⁶	2.7 x 10 ⁵	3.5 x 10 ⁷	1.7 x 10 ¹⁰	1.7 x 10 ⁵	
		Shale	6.4 x 10 ⁵	8.4 x 10 ⁵	1.9 x 10 ⁶	2.7 x 10 ⁵	3.5 x 10 ⁷	1.7 x 10 ¹⁰	1.8 x 10 ⁵	
		Basalt	8.9 x 10 ⁵	8.1 x 10 ⁵	1.7 x 10 ⁶	2.7 x 10 ⁵	3.5 x 10 ⁷	1.7 x 10 ¹⁰	1.8 x 10 ⁵	
2010	2030	Salt	1.2 x 10 ⁶	2.0 x 10 ⁶	1.9 x 10 ⁶	3.4 x 10 ⁵	3.4 x 10 ⁷	1.7 x 10 ¹⁰	2.2 x 10 ⁵	
		Granite	1.4 x 10 ⁶	2.0 x 10 ⁶	1.9 x 10 ⁶	3.8 x 10 ⁵	3.4 x 10 ⁷	1.8 x 10 ¹⁰	2.5 x 10 ⁵	
		Shale	1.1 x 10 ⁶	2.0 x 10 ⁶	2.0 x 10 ⁶	3.8 x 10 ⁵	3.4 x 10 ⁷	1.8 x 10 ¹⁰	2.6 x 10 ⁵	
		Basalt	1.3 x 10 ⁶	2.0 x 10 ⁶	1.9 x 10 ⁶	3.8 x 10 ⁵	3.4 x 10 ⁷	1.7 x 10 ¹⁰	1.8 x 10 ⁵	
250 GWe in 2000 and Steady State to 2040	2000	2000	Salt	8.4 x 10 ⁵	1.3 x 10 ⁶	1.8 x 10 ⁶	1.8 x 10 ⁵	4.5 x 10 ⁷	2.3 x 10 ¹⁰	2.1 x 10 ⁵
			Granite	1.1 x 10 ⁶	1.4 x 10 ⁶	1.8 x 10 ⁶	2.3 x 10 ⁵	4.5 x 10 ⁷	2.4 x 10 ¹⁰	2.5 x 10 ⁵
			Shale	7.2 x 10 ⁵	1.5 x 10 ⁶	2.0 x 10 ⁶	2.4 x 10 ⁵	4.5 x 10 ⁷	2.4 x 10 ¹⁰	2.6 x 10 ⁵
			Basalt	1.1 x 10 ⁶	1.4 x 10 ⁶	1.7 x 10 ⁶	2.3 x 10 ⁵	4.5 x 10 ⁷	2.3 x 10 ¹⁰	2.7 x 10 ⁵
	2000	2030	Salt	1.1 x 10 ⁶	1.5 x 10 ⁶	2.3 x 10 ⁶	3.1 x 10 ⁵	4.5 x 10 ⁷	2.3 x 10 ¹⁰	2.3 x 10 ⁵
			Granite	1.3 x 10 ⁶	1.6 x 10 ⁶	2.3 x 10 ⁶	3.4 x 10 ⁵	4.5 x 10 ⁷	2.4 x 10 ¹⁰	2.6 x 10 ⁵
			Shale	9.4 x 10 ⁵	1.6 x 10 ⁶	2.6 x 10 ⁶	3.6 x 10 ⁵	4.5 x 10 ⁷	2.4 x 10 ¹⁰	2.8 x 10 ⁵
			Basalt	1.3 x 10 ⁶	1.6 x 10 ⁶	2.2 x 10 ⁶	3.4 x 10 ⁵	4.5 x 10 ⁷	2.4 x 10 ¹⁰	2.8 x 10 ⁵
250 GWe in 2000 and 500 GWe in 2040	2000	2000	Salt	1.1 x 10 ⁶	1.4 x 10 ⁶	2.3 x 10 ⁶	2.3 x 10 ⁵	6.0 x 10 ⁷	3.0 x 10 ¹⁰	2.7 x 10 ⁵
			Granite	1.4 x 10 ⁶	1.6 x 10 ⁶	2.4 x 10 ⁶	2.9 x 10 ⁵	6.0 x 10 ⁷	3.2 x 10 ¹⁰	3.4 x 10 ⁵
			Shale	8.3 x 10 ⁵	1.6 x 10 ⁶	2.8 x 10 ⁶	3.0 x 10 ⁵	6.0 x 10 ⁷	3.2 x 10 ¹⁰	3.4 x 10 ⁵
			Basalt	1.4 x 10 ⁶	1.5 x 10 ⁶	2.3 x 10 ⁶	2.8 x 10 ⁵	6.0 x 10 ⁷	3.1 x 10 ¹⁰	3.5 x 10 ⁵
	2000	2020	Salt	1.3 x 10 ⁶	1.7 x 10 ⁶	3.2 x 10 ⁶	3.9 x 10 ⁵	6.0 x 10 ⁷	3.1 x 10 ¹⁰	2.9 x 10 ⁵
			Granite	1.6 x 10 ⁶	1.9 x 10 ⁶	3.2 x 10 ⁶	4.3 x 10 ⁵	6.0 x 10 ⁷	3.2 x 10 ¹⁰	3.4 x 10 ⁵
			Shale	1.2 x 10 ⁶	1.9 x 10 ⁶	3.5 x 10 ⁶	4.5 x 10 ⁵	6.0 x 10 ⁷	3.3 x 10 ¹⁰	3.6 x 10 ⁵
			Basalt	1.7 x 10 ⁶	1.8 x 10 ⁶	3.1 x 10 ⁶	4.4 x 10 ⁵	6.0 x 10 ⁷	3.2 x 10 ¹⁰	3.6 x 10 ⁵

TABLE A.6.3. Resource Commitments for Shipping Casks

<u>Growth Assumption</u>	<u>Reprocessing Startup Date</u>	<u>Repository Startup Date</u>	<u>Steel, MT</u>	<u>Lead, MT</u>
<u>Once-Through Cycle</u>				
Present inventory	NA ^(a)	1990	1.9×10^2	5.2×10^2
	NA	2010	1.9×10^2	5.2×10^2
	NA	2030	1.9×10^2	5.2×10^2
Present capacity to retirement	NA	1990	9.2×10^2	2.5×10^3
	NA	2010	1.3×10^3	3.3×10^3
	NA	2030	1.4×10^3	3.5×10^3
250 GWe in 2000 and decline to 0 in 2040	NA	1990	4.1×10^3	1.1×10^4
	NA	2010	6.2×10^3	1.6×10^4
	NA	2030	6.8×10^3	1.8×10^4
250 GWe in 2000 and steady state to 2040	NA	2000	6.7×10^3	1.8×10^4
	NA	2020	9.0×10^3	2.3×10^4
250 GWe in 2000 and 500 GWe in 2040	NA	2000	8.7×10^3	2.3×10^4
	NA	2040	1.2×10^4	3.1×10^4
Present capacity to retirement	NA	No action	7.0×10^2	1.9×10^3
250 GWe in 2000 and decline to 0 in 2040	NA	No action	3.6×10^3	9.4×10^3
<u>Reprocessing Cycles</u>				
250 GWe in 2000 and decline to 0 in 2040	1990	1990	6.1×10^3	1.7×10^4
	1990	2010	8.2×10^3	2.3×10^4
	2010	2010	8.1×10^3	2.2×10^4
	1990	2030	8.3×10^3	2.3×10^4
	2010	2030	1.0×10^4	2.8×10^4
250 GWe in 2000 and steady state to 2040	2000	2000	9.3×10^3	2.5×10^4
	2000	2020	1.2×10^4	3.4×10^4
250 GWe in 2000 and 500 GWe in 2040	2000	2000	1.2×10^4	3.3×10^4
	2000	2040	1.6×10^4	4.5×10^4

^(a) NA = not applicable.

A.7 TRANSPORTATION REQUIREMENTS

The transportation requirements tables (A.7.1 and A.7.2) show the number of shipments required by waste type, case and mode of transportation for both fuel cycles and for all cases analyzed.

TABLE A.7.1. Transportation Requirements Using the Once-Through Fuel Cycle

<u>Case</u>	<u>Growth Assumption</u>	<u>Repository Start-Up Date</u>	<u>Transport Mode</u>	<u>Spent Fuel Shipments (thousands)</u>
1	Present Inventory Only	None	Rail	0
			Truck	0
		1990	Rail	2.3
			Truck	2.3
		2010	Rail	2.3
			Truck	2.3
2030	Rail	2.3		
	Truck	2.3		
2	Present Capacity--Normal Life	None	Rail	8.4
			Truck	8.6
		1990	Rail	13.3
			Truck	11.1
		2010	Rail	18.0
			Truck	11.1
2030	Rail	19.0		
	Truck	11.1		
3	250 GWe System by Year 2000	None	Rail	45.0
			Truck	45.8
		1990	Rail	60.5
			Truck	55.6
		2010	Rail	88.6
			Truck	55.6
2030	Rail	95.7		
	Truck	55.6		
4	250 GWe System and Steady State	2000	Rail	96.6
			Truck	73.4
		2020	Rail	127.2
			Truck	73.4
5	500 GWe System by 2040	2000	Rail	125.7
			Truck	99.4
		2020	Rail	169.5
			Truck	99.4

TABLE A.7.2. Transportation Requirements Using the Reprocessing Cycle

Case	Growth Assumption	Reprocessing Start-Up Date	Repository Start-Up Date	Transport Mode	Thousands of Shipments						
					Spent Fuel	HLW	Canisters	RH-TRU Drums	CH-TRU Drums & Boxes	Decommissioning Waste	Total
1	Present Inventory Only	NA(a)	NA	NA	NA	NA	NA	NA	NA	NA	NA
2	Present Capacity--Normal Life	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA
3	250 GWe System by Year 2000	1990	1990	Rail	59	8.8	22	--	--	--	90
				Truck	56	--	--	108	21	4.7	190
		1990	2010	Rail	59	17	43	--	--	--	119
				Truck	56	--	--	213	40	4.7	314
		2010	2010	Rail	86	8.7	22	--	--	--	117
				Truck	56	--	--	108	14	3.7	182
		1990	2030	Rail	59	17	44	--	--	--	120
				Truck	56	--	--	215	41	4.7	317
2010	2030	Rail	86	17	44	--	--	--	147		
		Truck	56	--	--	215	29	3.7	304		
4	250 GWe System and Steady State	2000	2000	Rail	95	12	29	--	--	--	136
				Truck	73	--	--	142	33	2.4	250
		2000	2020	Rail	95	23	58	--	--	--	176
				Truck	73	--	--	284	53	2.4	412
5	500 GWe System by Year 2040	2000	2000	Rail	124	16	39	--	--	--	179
				Truck	99	--	--	192	50	2.4	343
		2000	2020	Rail	124	31	78	--	--	--	233
				Truck	99	--	--	385	80	2.4	566

(a) NA = not applicable.

A.8 SUPPLEMENTARY PREDISPOSAL COST DATA

The predisposal cost tables (A.8.1 through A.8.4) list the capital, operating, and levelized unit cost estimates for facilities for spent fuel treatment and storage, treatment of wastes from uranium and plutonium recycle, and interim storage of treated wastes. Costs for the example concepts (used in the system simulation cost determination) and for other optional methods are both shown. A table is also included showing capital costs for shipping casks and freight charge estimates for waste transportation over the generic distances used in this Statement.

TABLE A.8.1. Cost Estimates for Treatment and Storage of Spent Fuel

Activity	Total Capital Cost ^(a) \$10 ⁶	Annual Operation and Maintenance \$10 ⁶ /yr	Levelized Unit Cost	
			Private ^(b) Ownership \$/kg HM	Federal ^(b) Ownership \$/kg HM
Independent Unpackaged Water Basin Storage	234	5	212 ± 35%	117 ± 35%
Incremental 1000 MTHM Receiving Capacity at Above Facility	28	1.5	7 ± 40%	3.5 ± 35%
Spent Fuel Packaging Facility	128	13	29.6 ± 30%	18.30 ± 25%
Independent Spent Fuel Receiving Facility	92	1.5	---	9.5 ± 30%
Long-Term Packaged Spent Fuel Storage				
• Water Basin	296	392 ^(c)	---	38.40 ± 20%
• Air-Cooled Vault	595	1.3	---	34.90 ± 30%
• Dry Caisson	341	39 ^(d)	---	22.20 ± 25%
• Surface Cask	258	264 ^(e)	---	30.20 ± 20%

(a) Includes owner's costs.

(b) See DOE/ET-0028, Vol. 1, Section 3.8 for financial parameters relating to ownership.

(c) Includes \$389 million for incremental costs of using stainless steel canisters and storage racks.

(d) Include \$37 million for carbon steel storage casks

(e) Includes \$262 million for storage casks.

TABLE A.8.2. Cost Estimates for Treatment of Waste from Uranium and Plutonium Recycle

Activity	Total Capital Cost \$10 ⁶	Annual Operation and Maintenance \$10 ⁶ /yr	Levelized Unit Cost \$/kg HM ^(a)
5-Yr High-Level Liquid Waste Storage	282	6.6	42.00 ± 30%
High-Level Liquid Waste Solidification			
● Calcination	76	6.8	13.00 ± 35%
● Vitrification	55	7.1	10.40 ± 35%
Fuel Residue Packaging			
● Packaging Without Compaction	17	4.8	4.90 ± 25%
● Mechanical Compaction	20	3.5	4.60 ± 30%
● Melting	27	3.2	5.20 ± 35%
Failed Equipment and Non-Combustible Waste Packaging			
● At Reprocessing Plant	27	1.6	4.20 ± 55%
● At MOX Fuel Fabrication Plant	3.7	0.4	0.60 ± 55%
Combustible and Compactable Waste Treatment			
● At Reprocessing Plant			
- Incineration	16.9	1.6	3.40 ± 35%
- Package Only	18.1	0.8	2.30 ± 35%
● At MOX Fuel Fabrication Plant			
- Incineration	6.4	0.3	1.00 ± 35%
- Package Only	2.9	0.1	0.40 ± 35%
Degraded Solvent Treatment	8	0.1	1.40 ± 40%
Waste Immobilization			
● At Reprocessing Plant			
- In Bitumen	16	0.6	2.30 ± 35%
- In Cement	16	0.7	2.30 ± 35%
● At MOX Fuel Fabrication Plant			
- In Bitumen	14	0.3	1.40 ± 35%
- In Cement	13.5	0.3	1.40 ± 35%
Off-Gas Treatment			
● Iodine Recovery	12.8	0.8	2.00 ± 40%
● Carbon Recovery (w/o krypton recovery)	8.2	0.1	1.20 ± 40%
● Krypton Recovery (w/o carbon recovery)	25.8	1.3	4.00 ± 40%
● Combined Iodine, Carbon and Krypton Recovery	39.8	2.2	6.10 ± 40%
● Vessel Off-Gas Treatment	26.7	2.6	3.90 ± 35%
Off-Gas Filtration at Reprocessing Plant			
● Prefilters and HEPA Filters	11.7	0.6	1.80 ± 35%
● Sand Filter and HEPA Filters	28.1	0.6	3.80 ± 40%
● Deep-Bed Glass Filter and HEPA Filters	12.8	0.6	2.50 ± 40%

(a) Costs may be expressed in \$/GW-yr by multiplying by 38,000 kgHM/GW-yr

TABLE A.8.3. Cost Estimates for Interim Storage of Waste from Uranium and Plutonium Recycle

Activity	Total Capital Cost \$10 ⁶	Annual Operation and Maintenance \$10 ⁶ /yr	Levelized Unit Cost	
			Private Ownership \$/kg HM	Federal Ownership \$/kg HM
5-Yr Solidified High-Level Waste Basin Storage and Shipping Facility at Reprocessing Plant	99	3	13.80 ± 40%	---
Solidified High-Level Waste Storage Using the Sealed Cask Concept				
● HLW Accumulated to:				
- 1990	105	3.3	---	30.80 ± 70%
- 1995	115	8.5	---	15.80 ± 20%
- 2000	126	12.7	---	12.90 ± 20%
Fuel Residue Storage				
● 5-Yr Storage at Reprocessing Plant				
- Vault Concept	140	0.6	41.40 ± 25%	---
- Near-Surface Concept	41	0.3	12.30 ± 25%	---
● Storage to 1995 at Independent Site				
- Vault Concept	673	1.0	---	20.30 ± 25%
- Near-Surface Concept	191	0.9	---	6.20 ± 25%
TRU Intermediate-Level Waste Storage				
● 5-Yr Storage at Reprocessing Plant				
- Outdoor Subsurface Concept	45	0.2	9.30 ± 30%	---
- Indoor Shielded Concept	19	0.1	5.20 ± 30%	---
● Storage to 1995 at Independent Site				
- Outdoor Subsurface Concept	222	0.6	---	5.90 ± 30%
- Indoor Shielded Concept	87	0.4	---	2.60 ± 30%
TRU Low-Level Waste Storage				
● 5-Yr Storage at Reprocessing Plant				
- Outdoor Surface Concept	1.3	0.02	0.40 ± 30%	---
- Indoor Unshielded Concept	1.5	0.03	0.50 ± 25%	---
● 5-Yr Storage at MOX-FFP				
- Outdoor Surface Concept	1.2	0.02	0.40 ± 25%	---
- Indoor Unshielded Concept	1.2	0.02	0.40 ± 25%	---
● Storage to 1995 at Independent Site				
- Outdoor Surface Concept	6.4	0.1	---	0.30 ± 25%
- Indoor Unshielded Concept	10.7	0.1	---	0.40 ± 20%
Plutonium Oxide Storage ^(a)				
● 30 MT Facilities at Reprocessing Plant	281	4	33.70 ± 20%	---
● 200 MT Facility at Reprocessing Plant	494	3.2	50.00 ± 30%	---
● 200 MT Independent Site Facility				
- Accumulate to 1990	263	3.2	---	22.90 ± 25%
- Accumulate to 2000	1,053	6.7	---	22.50 ± 25%
Krypton Storage	192	0.2	16.40 ± 40%	---

(a) Plutonium oxide storage where plutonium is considered a waste is only needed in the event that spent fuel is reprocessed to recover the uranium value and remove plutonium.

TABLE A.8.4. Cost Estimates for Waste Transportation

Waste Shipment	Transport Mode	Cask Type	Single Cask Cost, \$10 ³	Distance, miles	Round Trip Freight Cost, \$10 ³	Unit Cost \$/kg HM
Unpackaged Spent Fuel	Rail	NLI 10/25	3,500	1,000	22	16.20
		IF-300	3,500	1,500	25	22.50
	Truck	NFS-4	1,050	1,000	3	18.50
				1,500	5	26.40
Packaged Spent Fuel	Rail	Modified NLI 10/24	3,500	1,500	25	32.00
Solidified High-Level Waste ^(a)	Rail	Conceptual	2,900	1,500	25	3.40
Fuel Residues						
Packaged Only	Rail	Conceptual	700	1,500	25	3.50
Mechanically Compacted	Rail	Conceptual	700	1,500	25	2.00
Melted	Rail	Conceptual	700	1,500	25	1.40
Non-High-Level TRU with a Surface Radiation Rate of						
0.2 R/hr	Truck	36 Drums ^(c)	100	1,500	3,200	0.24
				(1,000)	(2,300)	(0.21)
0.2 to 1.0 R/hr	Truck	36 Drums ^(c)	160	1,500	3,200	0.54
				(1,000)	(2,300)	(0.38)
1.0 to 1.0 R/hr	Truck	14 Drums ^(c)	140	1,500	3,200	0.30
				(1,000)	(2,300)	(0.21)
10 R/hr	Truck	6 Drums ^(c)	180	1,500	3,200	1.43
				(1,000)	(2,300)	(1.06)
Plutonium Oxide	Truck	PPP-1	260	1,500	16	0.80 ^(d)

(a) Costs of high-level waste transportation are about the same for calcined or vitrified waste.

(b) The costs shown in the table assume combustible waste is incinerated and all drummed waste is immobilized in cement.

(c) All casks are Type B casks. All casks are shielded except for the cask with drums measuring less than 0.2 R/hr. (DOE/ET-0028, Vol. 4, Section 6.6).

(d) Equivalent to about \$9 per gram of plutonium.

A.9 SUPPLEMENTARY SYSTEM COST DATA

The systems cost tables (A.9.1a through A.9.6) provide additional detail on the breakdown of power costs by major functions and the differences in power cost as influenced by repository media. Four sets of tables are included. The first two sets break down the costs by the functions of spent fuel storage and transport; spent fuel treatment; other waste treatment, storage and transport; disposal; and research and development for both fuel cycles. The latter two sets break down the total system costs by repository media for both fuel cycles. Each set consists of three tables with costs calculated at discount rates of 0, 7 and 10%. In addition to these tables, two tables are provided to display the estimated research and development costs (including site verification costs) for waste isolation.

TABLE A.9.1a. Allocation of Total-System Waste Management Unit Costs with the Once-Through Cycle Using a 0% Discount Rate, mills/kWh

<u>Case</u>	<u>Nuclear Power Growth Assumption</u>	<u>Repository Startup Date</u>	<u>Spent Fuel Storage and Transport</u>	<u>Spent Fuel Treatment</u>	<u>Disposal</u>	<u>Research and Development</u>	<u>Total</u>
1	Present Inventory Only	1990	0.63	0.10	0.51 to 0.62	1.6	2.9 to 3.0
		2010	1.31	0.10	0.51 to 0.62	2.3	4.2 to 4.3
		2030	1.98	0.10	0.51 to 0.62	5.0	7.6 to 7.7
		None	3.28			0.36	3.6
2	Present Capacity-- Normal Life	1990	0.42	0.08	0.22 to 0.37	0.26	1.0 to 1.1
		2010	0.79	0.08	0.22 to 0.37	0.36	1.5 to 1.6
		2030	0.92	0.08	0.22 to 0.37	0.79	2.0 to 2.2
		None	1.0			0.06	1.1
3	250 GWe system by Year 2000 and Normal Life	1990	0.33	0.08	0.22 to 0.37	0.06	0.69 to 0.84
		2010	0.69	0.08	0.22 to 0.37	0.08	1.1 to 1.2
		2030	0.84	0.08	0.22 to 0.37	0.17	1.3 to 1.5
		None	0.87			0.01	0.88
4	250 GWe System by Year 2000 and Steady State	2000	0.46	0.07	0.21 to 0.36	0.05	0.80 to 0.95
		2020	0.75	0.08	0.21 to 0.36	0.10	1.1 to 1.3
5	500 GWe System by Year 2040	2000	0.42	0.07	0.21 to 0.35	0.04	0.74 to 0.88
		2020	0.74	0.07	0.21 to 0.35	0.07	1.1 to 1.2

TABLE A.9.1b. Allocation of Total-System Waste Management Unit Costs with the Once-Through Cycle Using a 7% Discount Rate, mills/kWh

Case	Nuclear Power Growth Assumption	Repository Startup Date	Spent Fuel Storage and Transport	Spent Fuel Treatment	Disposal	Research and Development	Total
1	Present Inventory	1990	0.35	0.04	0.18 to 0.22	1.1	1.6 to 1.7
		2010	0.44	0.01	0.05 to 0.06	1.1	1.6
		2030	0.47	0.002	0.01	1.5	2.0
		None	0.48			0.30	.78
2	Present Capacity-- Normal Life	1990	0.34	0.03	0.10 to 0.17	0.37	0.85 to 0.92
		2010	0.45	0.01	0.03 to 0.04	0.39	0.87 to 0.89
		2030	0.45	0.0025	0.01	0.53	1.0
		None	0.45			0.11	0.56
3	250 GWe System by Year 2000 and Normal Life	1990	0.29	0.04	0.11 to 0.19	0.17	0.61 to 0.69
		2010	0.44	0.01	0.04 to 0.06	0.16	0.65 to 0.68
		2030	0.46	0.003	0.01 to 0.02	0.21	0.68
		None	0.45			0.04	0.49
4	250 GWe System by Year 2000 and Steady State	2000	0.41	0.02	0.07 to 0.12	0.16	0.66 to 0.71
		2020	0.45	0.01	0.02 to 0.03	0.19	0.67 to 0.69
5	500 GWe System by Year 2040	2000	0.40	0.03	0.07 to 0.12	0.14	0.64 to 0.69
		2020	0.46	0.01	0.02 to 0.03	0.17	0.66 to 0.67

TABLE A.9.1c. Allocation of Total-System Waste Management Unit Costs with the Once-Through Cycle Using a 10% Discount Rate, mills/kWh

<u>Case</u>	<u>Nuclear Power Growth Assumption</u>	<u>Repository Startup Date</u>	<u>Spent Fuel Storage and Transport</u>	<u>Spent Fuel Treatment</u>	<u>Disposal</u>	<u>Research and Development</u>	<u>Total</u>
1	Present Inventory Only	1990	0.28	0.02	0.12 to 0.15	0.90	1.3 to 1.4
		2010	0.32	0.004	0.02	0.87	1.2
		2030	0.33	0.0006	0.002 to 0.003	1.1	1.4
		None	0.33			0.28	0.61
2	Present Capacity-- Normal Life	1990	0.31	0.02	0.07 to 0.11	0.39	0.79 to 0.83
		2010	0.37	0.004	0.01 to 0.02	0.38	0.77
		2030	0.37	0.0005	0.0015 to 0.0026	0.47	0.85
		None	0.38			0.12	0.50
3	250 GWe System by Year 2000 and Normal Life	1990	0.28	0.03	0.08 to 0.14	0.21	0.59 to 0.65
		2010	0.37	0.01	0.02 to 0.03	0.19	0.58 to 0.59
		2030	0.38	0.0007	0.002 to 0.004	0.23	0.61
		None	0.38			0.06	0.44
4	250 GWe System by Year 2000 and Steady State	2000	0.36	0.01	0.04 to 0.06	0.20	0.61 to 0.63
		2020	0.38	0.002	0.01	0.22	0.60 to 0.61
5	500 GWe System by Year 2040	2000	0.36	0.01	0.04 to 0.07	0.19	0.60 to 0.62
		2020	0.38	0.002	0.01	0.21	0.59 to 0.60

TABLE A.9.2a. Allocation of Total-System Waste Management Unit Costs with the Reprocessing Cycle Using a 0% Discount Rate, mills/kWh

Case	Nuclear Power Growth Assumption	Repository Startup Date	Reprocessing Startup Date	Spent Fuel Storage and Transport	Waste Treatment Storage and Transport	Disposal	Research and Development	Total
1	Present Inventory Only	NA(a)	NA	NA	NA	NA	NA	NA
2	Present Capacity--Normal Life	NA	NA	NA	NA	NA	NA	NA
3	250 GWe System by Year 2000 and Normal Life	1990	1990	0.23	0.53 to 0.76	0.23 to 0.48	0.06	1.1 to 1.5
		2010	1990	0.24	0.51 to 0.58	0.21 to 0.36	0.08	1.0 to 1.3
		2010	2010	0.65	0.47 to 0.56	0.18 to 0.31	0.08	1.4 to 1.6
		2030	1990	0.24	0.51	0.19 to 0.30	0.17	1.1 to 1.2
		2030	2010	0.65	0.49 to 0.51	0.18 to 0.28	0.17	1.5 to 1.6
4	250 GWe System by Year 2000 and Steady State	2000	2000	0.40	0.49 to 0.61	0.20 to 0.35	0.05	1.1 to 1.4
		2020	2000	0.41	0.48 to 0.53	0.18 to 0.32	0.10	1.2 to 1.4
5	500 GWe System by Year 2040	2000	2000	0.37	0.48 to 0.62	0.20 to 0.36	0.04	1.1 to 1.4
		2020	2000	0.37	0.47 to 0.53	0.18 to 0.32	0.07	1.1 to 1.3

(a) NA = not applicable.

TABLE A.9.2b. Allocation of Total-System Waste Management Unit Costs with the Reprocessing Cycle Using a 7% Discount Rate, mills/kWh

Case	Nuclear Power Growth Assumption	Repository Startup Date	Reprocessing Startup Date	Spent Fuel Storage and Transport	Waste Treatment Storage and Transport	Disposal	Research and Development	Total
1	Present Inventory Only	NA(a)	NA	NA	NA	NA	NA	NA
2	Present Capacity--Normal Life	NA	NA	NA	NA	NA	NA	NA
3	250 GWe system by Year 2000 and Normal Life	1990	1990	0.25	0.25 to 0.32	0.09 to 0.18	0.17	0.76 to 0.91
		2010	1990	0.25	0.24	0.03 to 0.06	0.16	0.68 to 0.71
		2010	2010	0.43	0.07	0.02 to 0.04	0.16	0.68 to 0.70
		2030	1990	0.25	0.23	0.01 to 0.02	0.21	0.70
		2030	2010	0.43	0.06	0.01	0.21	0.71 to 0.72
4	250 GWe System by Year 2000 and Steady State	2000	2000	0.38	0.16	0.05 to 0.09	0.16	0.73 to 0.79
		2000	2020	0.39	0.13 to 0.14	0.02 to 0.03	0.19	0.73 to 0.74
5	500 GWe System by Year 2040	2000	2000	0.37	0.15 to 0.18	0.05 to 0.09	0.15	0.72 to 0.79
		2020	2000	0.37	0.14 to 0.15	0.02 to 0.04	0.17	0.71 to 0.73

(a) NA = not applicable.

TABLE A.9.2c. Allocation of Total-System Waste Management Unit Costs with the Reprocessing Cycle Using a 10% Discount Rate, mills/kWh

Case	Nuclear Power Growth Assumption	Repository Startup Date	Reprocessing Startup Date	Spent Fuel Storage and Transport	Waste Treatment Storage and Transport	Disposal	Research and Development	Total
1	Present Inventory Only	NA(a)	NA	NA	NA	NA	NA	NA
2	Present Capacity--Normal Life	NA	NA	NA	NA	NA	NA	NA
3	250 GWe System by Year 2000 and Normal Life	1990	1990	0.24	0.17 to 0.22	0.06 to 0.09	0.21	0.68 to 0.77
		2000	1990	0.24	0.16	0.01 to 0.02	0.19	0.60 to 0.62
		2010	2010	0.37	0.03	0.01	0.19	0.59 to 0.60
		2030	1990	0.24	0.16	0.002 to 0.004	0.23	0.63
		2030	2010	0.37	0.02	0.002 to 0.003	0.23	0.62
4	250 GWe System by Year 2000 and Steady State	2000	2000	0.34	0.07 to 0.08	0.02 to 0.04	0.20	0.63 to 0.66
		2020	2000	0.34	0.07	0.01	0.22	0.63 to 0.64
5	500 GWe System by Year 2040	2000	2000	0.34	0.08 to 0.09	0.03 to 0.05	0.19	0.63 to 0.66
		2020	2000	0.34	0.07	0.01	0.21	0.62 to 0.63

(a) NA = not applicable.

TABLE A.9.3a. Repository Media Effect on Total-System Waste Management Unit Cost with the Once-Through Cycle Using a 0% Discount Rate

Case	Nuclear Power Growth Assumption	Startup Date	mills/kWh			
			Salt Repository	Granite Repository	Shale Repository	Basalt Repository
1	Present Inventory Only	1990	2.85	2.91	2.86	2.97
		2010	4.19	4.25	4.20	4.31
		2030	7.61	7.67	7.61	7.72
2	Present Capacity--Normal Life	1990	0.98	1.09	1.00	1.13
		2010	1.45	1.56	1.47	1.60
		2030	2.01	2.13	2.04	2.17
3	250 GWe System by Year 2000 and Normal Life	1990	0.69	0.80	0.71	0.84
		2010	1.07	1.18	1.09	1.22
		2030	1.31	1.42	1.34	1.46
4	250 GWe System by Year 2000 and Steady State	2000	0.80	0.91	0.82	0.95
		2020	1.14	1.25	1.16	1.28
5	500 GWe System by Year 2040	2000	0.74	0.84	0.76	0.88
		2020	1.10	1.20	1.12	1.24

TABLE A.9.3b. Repository Media Effect on Total-System Waste Management Unit Cost with the Once-Through Cycle Using a 7% Discount Rate

Case	Nuclear Power Growth Assumption	Startup Date	mills/kWh			
			Salt Repository	Granite Repository	Shale Repository	Basalt Repository
1	Present Inventory Only	1990	1.62	1.64	1.62	1.66
		2010	1.59	1.59	1.59	1.60
		2030	1.97	1.97	1.97	1.98
2	Present Capacity--Normal Life	1990	0.85	0.90	0.86	0.92
		2010	0.87	0.88	0.87	0.89
		2030	0.99	0.99	0.99	1.00
3	250 GWe System by Year 2000 and Normal Life	1990	0.61	0.67	0.63	0.69
		2010	0.65	0.67	0.66	0.68
		2030	0.68	0.68	0.68	0.68
4	250 GWe System by Year 2000 and Steady State	2000	0.66	0.70	0.67	0.71
		2020	0.67	0.68	0.68	0.69
5	500 GWe system by Year 2040	2000	0.64	0.68	0.65	0.69
		2020	0.66	0.67	0.66	0.67

TABLE A.9.3c. Repository Media Effect on Total-System Waste Management Unit Cost with the Once-Through Cycle Using a 10% Discount Rate

Case	Nuclear Power Growth Assumption	Startup Date	mills/kWh			
			Salt Repository	Granite Repository	Shale Repository	Basalt Repository
1	Present Inventory Only	1990	1.32	1.33	1.32	1.35
		2010	1.22	1.22	1.22	1.22
		2030	1.42	1.42	1.42	1.42
2	Present Capacity--Normal Life	1990	0.79	0.82	0.79	0.83
		2010	0.77	0.77	0.77	0.77
		2030	0.85	0.85	0.85	0.85
3	250 GWe System by Year 2000 and Normal Life	1990	0.59	0.63	0.60	0.65
		2010	0.58	0.59	0.58	0.59
		2030	0.61	0.61	0.61	0.61
4	250 GWe System by Year 2000 and Steady State	2000	0.61	0.63	0.61	0.63
		2020	0.60	0.61	0.61	0.61
5	500 GWe System by Year 2040	2000	0.60	0.62	0.60	0.62
		2020	0.59	0.60	0.60	0.60

TABLE A.9.4a. Repository Media Effect on Total-System Waste Management Unit Costs^(a) with the Reprocessing Cycle Using a 0% Discount Rate

Case	Nuclear Power Growth Assumption	Reprocessing Startup	Repository Startup	mills/kWh			
				Salt Repository	Granite Repository	Shale Repository	Basalt Repository
1	Present Inventory Only	NA ^(b)	NA	NA	NA	NA	NA
2	Present Capacity--Normal Life	NA	NA	NA	NA	NA	NA
3	250 GWe System by Year 2000 and Normal Life	1990	1990	1.06	1.34	1.51	1.54
		1990	2010	1.04	1.21	1.21	1.26
		2010	2010	1.39	1.53	1.55	1.61
		1990	2030	1.10	1.20	1.16	1.22
		2010	2030	1.49	1.58	1.54	1.61
4	250 GWe System by Year 2000 and Steady State	2000	2000	1.14	1.36	1.38	1.42
		2000	2020	1.17	1.31	1.30	1.36
5	500 GWe System by Year 2040	2000	2000	1.08	1.31	1.35	1.38
		2000	2020	1.09	1.24	1.23	1.29

(a) Includes spent fuel handling and storage.

(b) NA = not applicable.

TABLE A.9.4b. Repository Media Effect on Total-System Waste Management Unit Costs^(a) with the Reprocessing Cycle Using a 7% Discount Rate

Case	Nuclear Power Growth Assumption	Reprocessing Startup	Repository Startup	mills/kWh			
				Salt Repository	Granite Repository	Shale Repository	Basalt Repository
1	Present Inventory Only	NA ^(b)	NA	NA	NA	NA	NA
2	Present Capacity--Normal Life	NA	NA	NA	NA	NA	NA
3	250 GWe System by Year 2000 and Normal Life	1990	1990	0.76	0.86	0.89	0.91
		1990	2010	0.68	0.70	0.70	0.71
		2010	2010	0.68	0.69	0.69	0.70
		1990	2030	0.70	0.70	0.70	0.70
		2010	2030	0.71	0.72	0.72	0.72
4	250 GWe System by Year 2000 and Steady State	2000	2000	0.73	0.78	0.78	0.79
		2000	2020	0.73	0.74	0.74	0.74
5	500 GWe System by Year 2040	2000	2000	0.72	0.77	0.77	0.79
		2000	2020	0.71	0.72	0.73	0.73

(a) Includes spent fuel handling and storage.

(b) NA = not applicable.

TABLE A.9.4c. Repository Media Effect on Total-System Waste Management Unit Costs(a) with the Reprocessing Cycle Using a 10% Discount Rate

Case	Nuclear Power Growth Assumption	Reprocessing Startup	Repository Startup	mills/kWh			
				Salt Repository	Granite Repository	Shale Repository	Basalt Repository
1	Present Inventory Only	NA(b)	NA	NA	NA	NA	NA
2	Present Capacity--Normal Life	NA	NA	NA	NA	NA	NA
3	250 GWe System by Year 2000 and Normal Life	1990	1990	0.68	0.74	0.76	0.77
		1990	2010	0.60	0.61	0.61	0.62
		2010	2010	0.59	0.59	0.59	0.60
		1990	2030	0.63	0.63	0.63	0.63
		2010	2030	0.62	0.62	0.62	0.62
4	250 GWe System by Year 2000 and Steady State	2000	2000	0.63	0.66	0.66	0.66
		2000	2020	0.63	0.64	0.64	0.64
5	500 GWe System by Year 2040	2000	2000	0.63	0.65	0.65	0.66
		2000	2020	0.62	0.63	0.63	0.63

(a) Includes spent fuel handling and storage.

(b) NA = not applicable.

TABLE A.9.5. Estimated Research and Development Costs for Predisposal Management for a 1990 Repository Start, \$ Millions (a)

	<u>Spent Fuel Storage</u>	<u>Waste Treatment and Packaging</u>	<u>Waste Transport</u>	<u>Total</u>
1980 Cumulative	35	105	5	145
81	20	30	1	51
82	20	30	2	52
83	20	30	3	53
84		30	3	33
85		30	3	33
86		30	3	33
87		30	2	32
88		30	2	32
89		30	2	32
90		30	2	32
91		25	2	27
92		20	2	22
93		15	1.5	16.5
94		10	1	11
95		5	0.5	5.5
	<u>95</u>	<u>480</u>	<u>35</u>	<u>610</u>

(a) For later repository start up assumptions the predisposal R & D costs are extended for longer periods as shown in Table A.9.6.

TABLE A.9.6. Estimated Research and Development Cost (including site verification) for Waste Isolation

Cumulative	1990 Disposal Start			2000 Disposal Start			2010 Disposal Start			2020 Disposal Start	2030 Disposal Start
	Predisposal	Disposal ^(a)	Total	Predisposal	Disposal ^(b)	Total	Predisposal	Disposal ^(c)	Total		
1980	145	437	582	145	437	582	145	437	582	582	582
1981	51	161	212	51	158	189	51	138	189	190	190
1982	52	241	293	52	184	236	52	184	236	190	190
1983	53	257	310	53	243	296	53	201	254	190	190
1984	33	251	284	33	239	272	33	198	231	190	190
1985	33	232	265	33	212	245	33	187	220	190	190
1986	33	212	245	33	177	210	33	152	185	190	190
1987	32	123	155	32	72	104	32	59	91	190	190
1988	32	139	171	32	70	102	32	50	82	190	190
1989	32	138	170	32	78	110	32	46	78	190	190
1990	32	130	162	32	97	129	32	51	83	190	190
1991	27	59	86	32	105	137	32	50	82	190	190
1992	22	65	87	32	103	135	32	43	75	190	190
1993	17	91	108	32	102	134	32	48	80	190	190
1994	11	91	102	32	97	129	32	48	80	190	190
1995	6	78	84	32	105	137	32	48	80	190	190
1996		41	41	27	105	132	32	48	80	190	190
1997		41	41	22	97	119	32	43	75	190	190
1998		53	53	17	77	94	32	51	83	190	190
1999		52	52	11	90	101	32	58	90	190	190
2000		39	39	6	82	88	32	58	90	190	190
2001					43	43	27	63	90	130	190
2002					45	45	22	63	85	130	190
2003					45	45	17	86	103	130	190
2004					53	53	11	99	109	130	190
2005					52	52	6	97	103	130	190
2006					39	39		92	92	130	190
2007								82	82	130	190
2008								96	96	130	190
2009								106	106	130	190
2010								92	92	130	190
2011								60	60	130	130
2012								66	66	130	130
2013								92	92	130	130
2014								90	90	130	130
2015								91	91	130	130
2016								41	41	130	130
2017								41	41	130	130
2018								53	53	130	130
2019								52	52	130	130
2020								39	39	130	130
2021										65	130
2022										65	130
2023										65	130
2024										65	130
2025										65	130
2026										65	130
2027										65	130
2028										65	130
2029										65	130
2030										65	130
2031											65
2032											65
2033											65
2034											65
2035											65
2036											65
2037											65
2038											65
2039											65
2040											65
TOTALS	611	2931	3542	771	3187	3958	931	3698	4629	7632	9532

(a) Includes \$6.5 million/yr through 1993 for alternative disposal technologies.
 (b) Includes \$6.5 million/yr through 2003 for alternative disposal technologies.
 (c) Includes \$6.5 million/yr through 2013 for alternative disposal technologies.

A.10 SYSTEM REPOSITORY REQUIREMENTS

Tables A.10.1 and A.10.2 provide a complete listing of the calculated number of repositories required for each of system simulation cases for the once-through and reprocessing cycles, respectively.

TABLE A.10.1. Repository Requirements for Once-Through Cycle

Case	Nuclear Power Growth Assumption	Startup Date	Number of Repositories			
			Salt Repository	Granite Repository	Shale Repository	Basalt Repository
1	Present Inventory Only	1990	0.1	0.05	0.1	0.05
		2010	0.1	0.04	0.1	0.04
		2030	0.1	0.03	0.1	0.03
2	Present Capacity--Normal Life	1990	0.7	0.3	0.5	0.3
		2010	0.7	0.2	0.5	0.2
		2030	0.7	0.2	0.4	0.2
3	250 GWe System by Year 2000 and Normal Life	1990	3.5	1.5	2.6	1.5
		2010	3.6	1.3	2.4	1.3
		2030	3.5	1.0	2.1	1.0
4	250 GWe System by Year 2000 and Steady State	2000	4.7	1.9	3.4	1.9
		2020	4.8	1.6	3.0	1.6
5	500 GWe System by Year 2040	2000	6.3	2.6	4.6	2.6
		2020	6.5	2.1	4.1	2.1

TABLE A.10.2. Repository Requirements for Recycle Cases

Case	Nuclear Power Growth Assumption	Reprocessing Startup	Repository Startup	Number of Repositories			
				Salt Repository	Granite Repository	Shale Repository	Basalt Repository
1	Present Inventory Only	NA ^(a)	NA	NA	NA	NA	NA
2	Present Capacity--Normal Life	NA	NA	NA	NA	NA	NA
3	250 GWe System by Year 2000 and Normal Life	1990	1990	3.1	2.9	5.4	3.1
		1990	2010	2.8	2.4	4.7	2.6
		2010	2010	2.3	2.1	4.0	2.2
		1990	2030	2.3	2.0	3.9	2.2
		2010	2030	2.1	1.9	3.7	2.1
4	250 GWe System by Year 2000 and Steady State	2000	2000	3.6	3.3	6.3	3.5
		2000	2020	3.3	2.9	5.7	3.2
5	500 GWe System by Year 2040	2000	2000	5.0	4.6	8.7	4.9
		2000	2020	4.6	4.1	8.0	4.5

(a) Not Applicable.

APPENDIX B

GEOLOGIC DISPOSAL SUPPLEMENTARY INFORMATION

Factors relevant to selection of a geologic repository include the depth of the repository; the size and properties of waste form and host rock; seismic, tectonic and magnetic characteristics of the proposed repository; the hydrologic system and material resources near the repository; and the use of multiple geologic barriers. These factors will be considered in a three-stage site selection process.

B.1 DEPTH OF REPOSITORY

The optimum depth of the waste emplacement zone is a function of the geologic media and is thus site specific. A depth of 600 m is frequently cited because it is proposed depth for a test facility in salt in New Mexico (Claiborne and Gera 1974). A 1000 m depth has frequently been mentioned in the literature. The repository must be deep enough to rule out any significant effects from disruption by surface phenomena and to provide as long a pathway to man's environment as possible. Because of the variety of geologic media and settings in the United States, it should be possible to find a number of sites having appropriate host rock at suitable depths.

Because destructive natural surficial processes (for example, erosion, climate and weathering) may reduce the depth to the repository, the host rock should be deep enough to separate the repository from these processes and thus maintain geologic isolation. Baseline data to evaluate these factors can be obtained from historic and geologic evidence. Climatic conditions and associated erosional and weathering processes have an influence to variable depths, depending upon local conditions.

Climate and rock properties provide the conditions for erosion and weathering. The energy for transport of earth materials is provided by running water, moving ice, wind, and gravity. Records of present and paleoclimatic conditions must be evaluated to predict future climatic variations and to estimate possible depths of erosion. Typical climatic and related factors to be evaluated at a repository site include:

- daily and seasonal atmospheric conditions
- latitude and longitude
- altitude
- position with respect to ocean and/or global wind circulation patterns.

These four parameters are basic data required to establish the types of weathering forces and erosion that will act to reduce depth. For example, a high-latitude site and a possible past history of glaciation at these latitudes indicate a potential for glacial erosion.

The repository site can be characterized by its topography (land-surface configuration), unconsolidated surficial materials (soil), and underlying rock. Earth materials surrounding the repository are the prime barriers to movement of radioactive waste to the

biosphere. These earth material properties must be known in order to determine rates of erosion. The properties of these materials that relate to erosional processes are the strength, hardness, chemical composition, consistency, uniformity, and homogeneity.

Topography (land-surface configuration) has an economic impact because of its influence on ease of access for materials and transportation, the amount of surface modification required for construction of facilities such as buildings and railroads, and any unique problems such as landslide potential or flash flooding. In addition, steep terrain frequently indicates tectonic instability. In general, a relatively flat and open area with low relief is considered desirable.

Weathering is the chemical and physical decomposition and/or transport of surface and near-surface earth materials by surface erosional processes. It can decompose earth materials into smaller components that are more easily carried and deposited by other erosional processes. The weathering process can break down earth materials as deep as several hundred feet. In addition to climatic forces, the rate of weathering depends also on the resistance of earth materials to chemical deterioration and physical pressures. The major chemical and physical weathering processes are freezing and thawing, hydration, hydrolysis, oxidation, carbonation, dissolution, and expansion caused by unloading, crystal growth, thermal difference, and organic activity. All of these may remove material and thus decrease the depth to the isolated waste.

Water (stream) erosion processes are a function of a base level (Office of Waste Isolation 1977). Base level is a surface below which moving water cannot erode. The ultimate base level for stream erosion is generally considered to be sea level. Base level can change, however, over geologic time; for example, large fluctuations of sea level can occur during glacial periods. The mechanisms of a stream erosion are acquisition of weathered earth materials, abrasion of material through particle impact, transit abrasion of materials, and transport by the traction, suspension, or solution of weathered rock debris.

Erosional processes unaffected by a base level are those related to ice, wind, and gravity. These processes are important because of their potential for eroding below base level.

Erosion by ice is caused by glaciation, and the continental type has the greatest potential impact on depth of isolation (Office of Waste Isolation 1977). Glaciers are a dynamic mass of recrystallized snow and ice, and the character and longevity of a glacier depend on climatic factors. Glacial action alters the land surface and could reduce the depth of a repository by 1) plowing or scraping earth materials from a site, 2) abrasion of intact rock, and 3) assimilation of plowed and abraded material into the ice mass (Verhoogen et al. 1970). The depth to the repository may be effectively reduced if fracturing or faulting results from the loading and unloading of the ice on the land surface. Parameters affecting glacial erosion are ice temperature and thickness, earth material and structure, and topography. The depth below base level at which glaciers may erode can be substantial. The lower depth of glacial erosion at a repository site can be predicted to some extent from the glacial history.

Erosion by wind energy is a mechanical process. It requires the environmental conditions of no vegetation and uncemented dry earth materials (Verhoogen et al. 1970). These conditions are most prevalent in desert environments. Depth of possible wind erosion is controlled by wind velocity, duration, and other climatic conditions (Office of Waste Isolation 1977).

Mass-wasting, or gravitational erosion, is the movement of earth materials by gravity independent of water, glacier, or wind. The significance of mass-wasting is that it affects the whole body of the earth material and is not confined to a land environment. Mass-wasting occurs when the force of gravity on a mass of earth material exceeds the cohesive strength between the individual earth particles. Environmental components important to the mass-wasting process are weathering, geomorphology (topography), processes of stream, glacial, and wind erosion and sometimes earthquakes (Claiborne and Gera 1974).

Surface geologic processes cause the transport of earth materials to sites of deposition. Rates of deposition may be as imperceptibly slow as rates of erosion. However, they also may be significant over hundreds of thousands of years. Agents of deposition that should be evaluated for candidate repository site regions include runoff and streams, wind, glacial processes, and volcanism. A surface environment conducive to long-term deposition is somewhat favorable to repository containment because as the depth of sedimentary cover continually and gradually increases, so would the depth of burial.

B.2 DIMENSIONS AND PROPERTIES OF HOST ROCKS AND MEDIA

The host rock must have the properties and dimensions to assure geologic isolation (Office of Waste Isolation 1977). One method for defining the required dimensions of a repository medium is use of an "equilibrium release fringe concept." The concept assumes that the repository system contains the waste within a known or definable zone for the necessary time period. After a period of time, the competing factors of radioactive decay and chemical migration processes will produce an equilibrium zone or fringe that will not move or will move so slowly as to be insignificant. Using these definitions, a three-dimensional zone consisting of host rock material, repository and waste, is defined on the basis of host rock and waste package properties beyond which no waste or activity beyond a specified range is expected to migrate for the necessary time period. The specified range lies between the values for radioisotope concentration at the maximum natural concentration found in the world and the average U.S. natural background concentration. This condition is defined as an equilibrium condition, i.e., any material or activity released beyond the fringe or boundaries of the zone would be within the range of that which occurs naturally.

The size of the zone of effect will probably change throughout the repository's history. After sealing, the zone will be very nearly the size of the repository and the fringe will be located by radiation effects. At a later time in the repository's history, when the canisters and overpack material may have lost their integrity as barriers, the waste will be partially in contact with the host rock. The waste may then move slowly into the host rock by diffusion, concentration gradients or whatever forces are present to move it. The fringe bounding the zone of effects will expand as the zone slowly moves out from the repository. The size of the zone of effect and the location of the equilibrium fringe will depend on the host rock properties, the form of the waste, the activity and thermal state of the waste at the time the canisters became ineffective as containment, and other factors such as presence of water.

The location of the equilibrium release fringe is difficult to predict, particularly over time periods greater than several thousand years. Simulation by modeling may furnish some estimates if the necessary input data are available. The modeling would proceed under the assumption that no intrusions or disruptions occurred.

The required dimensions of the host rock relate closely to the radius of equilibrium release and are established as a function of the medium's properties and of engineering design of the repository. Important media properties that affect the radius of equilibrium release can be classified as thermal, chemical, and hydrologic.

The host rock dimensions must be large enough with respect to the repository dimensions to adequately disperse or contain all of the perturbations and loads induced by the repository. These dimensions will depend directly on site-specific geologic properties of the host rock. The host rock must also be of sufficient thickness to ensure that excavation and construction can proceed on several depth levels and over many acres of lateral extent. Adequate thickness of the zone adds assurance that the specific medium is of sufficient mass and extent to contain the waste and buffer the repository from materials with different

properties. The emplacement medium should be homogeneous and uniform in properties and composition, and the medium should extend some distance from the repository so that the response to the waste will be similar and more predictable. The concept of equilibrium-release radius can be used to derive the required host rock dimensions as discussed above. This is a consideration for modeling specific sites in the last stage of site selection.

Several engineered barriers will be built into repository design; however, they will probably have negligible permanence compared to the lifetime of the repository. The primary geologic barrier to waste migration will be the repository host rock itself. The effectiveness of the barrier will depend on the responses of the host rock to long-term effects of heating and irradiation. Rock response over the full range of expected repository conditions is not adequately understood; however, uncertainties can be overcome by more conservative design for waste emplacement.

Preliminary thermal loading analyses indicate that tensile forces will be induced near the outer margins of the repository (Office of Waste Isolation 1978f). Thus thermal expansion could create potential pathways for waste migration by fracturing or by opening pre-existing fractures. For salt strata this is not the problem; salt is expected to deform plastically and heal internal fractures. However, if the surrounding strata were breached by fracturing, salt could be vulnerable to rapid solution by ground water. Therefore, thermally induced permeability appears to be an important consideration for all host rock media.

Dip, inclination, or attitude of the units in the rock column or section is considered both from a construction standpoint and as indicators of past geologic stability. Flat or nearly horizontal units will probably be easier to tunnel through, mine and support if needed. Steeply dipping or inclined units, in general, indicate past deformation or movement and would likely be avoided if other areas can be found. Any geologic section with units of different inclinations or dip within the rock column may indicate the presence of erosion or weathering surfaces that might be selectively weak or permeable. Low and fairly uniform inclination or dips are probably most desirable.

Joints, fractures and faults are generally not favorable from a geologic site-selection point of view. They represent zones of weakness, movement, possible conduits for fluids and regions of anomalous properties compared to the general rock mass. They also increase the time and cost of investigations and complicate the modeling necessary for design. The presence of these features does not necessarily exclude a site; joints and fractures may be closed or sealed by mineral deposition and would not act as conduits and may be barriers to flow, and some faults can be shown to have had no movement for millions of years. However, in selecting general site areas risks and benefits of areas exhibiting these features need to be carefully considered.

A comparative survey of rock properties is included in Table B.2.1. Rock behavior and strength properties strongly affect design and underground construction. These aspects are discussed in following sections of this report.

TABLE B.2.1. Physical Properties of Media

Type of Properties	Parameter ^(a)	Salt	Granite	Shale	Basalt
Index	Unit weight, lb/ft ³ (density)	130 to 152	144 to 190	117 to 188	180
	Natural moisture content, %	0 to 1.1	0 to 0.32	0 to 38	nil
Stress-Strain	Young's modulus, lb/in. ²	0.09 x 10 ⁶ to 7.25 x 10 ⁶	2.3 x 10 ⁶ to 12.1 x 10 ⁶	2 x 10 ³ to 26.4 x 10 ⁶	1.8 x 10 ⁶
	Poisson's ratio	0.22 x to 0.50	0.045 to 0.39	0.03 to 0.50	0.26
Strength	Cohesion (1,500 to 3,500 psi ranges), lb/in.	900 to 1,700	--	0 to 4,250	--
	Friction angle,	20 to 36	--	4.2 to 56	--
	Uniaxial compressive strength, lb/in. ²	2,300 to 7,250	5,100 to 51,200	70 to 37,000	18,000 to 40,000
	Tensile strength, lb/in. ²	120 to 458	500 to 8,100	0 to 1,540	1,800 to 3,500
Thermal	Coefficient of linear thermal expansion, F-1	2.1 x 10 ⁻⁵	3.0 x 10 ⁻⁶ 6.0 x 10 ⁻⁶	4 x 10 ⁻⁶	3.0 x 10 ⁻⁶
	Heat capacity, Btu/lb- F	0.19 to 47.00	0.16 to 0.33	0.20	0.17 to 0.23
	Thermal conductivity, Btu/hr-ft- F	at 32 F-3.5 at 752 F-1.2	at 32 F-1.65 at 752 F-1.24	at 32 F-1.1 at 752 F-0.8	at 32 F-0.65 at 752 F-0.85
Hydrologic	Permeability, ft/yr	1.7 x 10 ⁻¹⁵ to 1 x 10 ⁻²	Very low if no joints or fractures	horizontal 10 to 10 vertical 1/2 to 1/10 times horizontal	Very low if unfractured and not jointed
	Porosity, %	1.4 to 10.0	0.5	0 to 45	0.6

(a) In English units.

Rocks are named and described according to their texture and mineralogy. However, their overall behavior may depend on details of petrography such as mineral composition, rock fabric, fluid inclusions, exotic mineral accumulations in joints and fractures, and trace element chemistry.

Petrography is important in determining the suitability of the host rock. These data will be collected and evaluated for specific sites during the site selection process. The basic properties of rock fabric and composition are discussed in Section 5.1.

Many chemical interactions are possible among mineral and fluid phases of the host rock, ground water, metal canisters, backfill material, and waste. The range of possible chemical interactions is described in Section 5.1. However, additional study of geochemical aspects is warranted.

Thermal properties of high diffusivity and conductivity and low thermal expansion are normally considered to be desirable. These properties result in maintaining lower waste temperatures and minimizing mechanical deformation (expansion). Design of a repository should restrict thermal loading so that excessive thermal expansion does not fracture the host rock and thereby increase permeability.

Chemical properties of different host rocks vary greatly, and the range of possible chemical reactions both before and after the containers may be breached may be significant. Favorable reactions between waste and the host rock include formation of insoluble radioactive compounds, formation of compounds containing water (thus reducing the quantity of free water), and sorption of radionuclides by the host rock. However, if corrosive fluids are produced by heating of the host rock, they may attack the canisters and result in early loss of this barrier. Chemical reactions between waste and host rock that form highly soluble or low melting-point compounds would be unfavorable. Chemical reactions can affect chemical transport by changing the composition and quantity of fluids and by changing the ionic strength of these fluids. Chemical reactions can produce liquids and gases under high pressure, and can change pH, Eh, viscosity, or density. Such factors can affect rock strength and rate of physical or chemical decomposition of the host rock.

Permeability is an important hydrologic property of the host rock and must be known to determine the rate of migration of radionuclides toward the biosphere. Very low permeability implies a relatively small radius of equilibrium release if other considerations are also favorable.

Properties of the medium also affect repository capacity and waste placement geometry. For example, low thermal conductivity of the host rock would require lower waste loading or greater spacing between canisters to maintain acceptable repository temperatures.

The total system of waste form, repository, surrounding geologic environment, and effects of waste disposal must be considered to identify any possible site-specific determinants of the radius of equilibrium release. If possible, the host rock dimensions should encompass the radius of equilibrium release. However, the radius may extend beyond the host rock and isolation of waste will be achieved by additional barriers in accordance with the multibarrier concept.

B.3 SEISMIC, TECTONIC AND MAGNETIC CONSIDERATIONS

The tectonic stability of the repository site must be sufficient to assure geologic isolation (Office of Waste Isolation 1977). Tectonics refers to the deformation of the lithosphere (the solid, near-earth-surface materials) caused by large-scale and local dynamic earth processes.

Tectonics, seismicity and volcanism relate to the stability of an area and reflect the past geological activity. Active or capable faults, a history of earthquakes and volcanism should not condemn an area if investigation can show that the activity was in the remote past (million to hundreds of million years ago) and has not occurred since. For preliminary selection of areas, crustal plate boundaries, areas of known active faults, and zones of recent earthquake and volcanic activity would be avoided.

Deformation of the lithosphere (tectonism) and the upward intrusion (or extrusion) of molten rock (magma) are important in site selection. Deformation of the crust may consist of folding, faulting, uplift, depression or diapirism. (A diapir is a fold in which the mobile core is injected into the overlying materials.) These processes, even though they may not directly disturb a repository site by fault displacement or venting of volcanic material, can significantly affect the regional hydrology over a hundred thousand years or more by altering the topography and the subsurface fluid flow. In this respect, magmatism and tectonism rank with climatic change as important factors in determining the evolution of the hydrologic environment. In selecting a site, optimal conditions of tectonic stability should be realized so that magmatism and tectonism will not adversely affect the hydrologic conditions at the site. To determine that only the site itself would not be directly disrupted by faulting or volcanism is not sufficient; the general region must be considered. In general, the tectonic constraints on site selection will generally be more difficult to satisfy for sites in the western U.S. than for sites east of the Rocky Mountains.

The theory of plate tectonics on a continental scale is believed appropriate for identifying areas of optimum tectonic stability in order to assess the constraints on site selection imposed by volcanic activity, tectonism, and seismicity. The plate tectonics theory explains in general the present global distribution of lithospheric deformation, magmatic activity and seismic activity, and also the geologic record of lithospheric deformation and magmatism over at least the past several hundred million years. Because volcanism has in general occurred in regions of crustal plate boundaries, some aspects of the evolution of the lithosphere during the next million years can be forecast quantitatively from plate tectonics; for example, it is possible to forecast, within a factor of about two, an increase of 50 km in the horizontal displacement across the San Andreas fault system in California. However, many important aspects of the evolution of the lithosphere can be forecast only in qualitative terms, if at all; for example, the effects of tectonism on the physiography of the Rio Grande Rift in New Mexico are difficult, if not impossible to forecast.

Through isotopic ratio dating, particularly for pre-Paleozoic rocks (older than 600 million years), former zones of crustal activity or mobility have been defined. Knowledge of these zones has proved particularly useful in studies of the early history of the North American continent. The question of renewed or future activity at these zones is debatable. Some of the former mobile zones have stabilized to form areas like the Canadian Shield (Dott and Batten 1971), and their relationship, if any, to present crustal plate boundaries is not clear. The plate tectonic theory was formulated a decade ago as basically a kinetic theory, and is now in the early stages of development into a full physical and chemical theory, incorporating geologic knowledge acquired over the past centuries. The driving mechanisms for plate tectonics are not presently understood.

The geologic stability, over the past 100 million years or more, of the major part of the U.S. east of the Rocky Mountains is readily explained in the framework of the plate tectonic theory. In this region the lithosphere has behaved essentially as a rigid plate, undergoing rigid-body rotation away from the Mid-Atlantic Ridge. The broad features of the present-day tectonics of the western U.S. arose, after episodes of continental accretion associated with consumption of oceanic lithosphere along the western margin of the continent, when the North American plate overrode an oceanic rise system, the remnants of which (the Juan de Fuca Ridge, the Gorda Ridge, and the ridge system in the Gulf of California) continue to create new oceanic lithosphere. Remaining to be explained is the relation between these events and the incipient continental rifting represented by the Snake River and Yellowstone volcanism and the Rio Grande Rift.

Consideration of the optimal region or regions of tectonic stability for siting purposes proceeds from the continental scale to regional and local scales, to ensure that sites are viewed in their proper context. Simple projection into the future from local geologic history alone is not a satisfactory basis for repository site selection. On the regional and local scales, site selection will necessarily involve uncertain projections from the geologic record. These projections will tend to be more tenuous in the more tectonically active regions. At the same time, in the less active regions the tectonic regime may be more difficult to ascertain because of fewer opportunities for the study of seismotectonics (the inference of the geometry of tectonic stress and faulting from earthquake mechanism determinations). In the span of a hundred thousand years or more, significant aseismic deformation may occur. Even in the relatively stable eastern U.S., local vertical surface velocities of a millimeter or more per year are ubiquitous. Motions of this magnitude, persisting over a period of hundreds of thousands or millions of years as in the case of the uplift of the Adirondacks, could result in erosion of hundreds of meters of overburden. It is uncertain whether such movements could present a serious problem for waste isolation even if they were not anticipated.

Tectonic activity varies in intensity throughout different regions of the North American continent. Most of the intense tectonic activity and virtually all the volcanic activity of the North American continent occur along the crustal plate boundaries. A repository site will be located in a relatively stable tectonic region. In general, the underground

parts of the repository are not expected to be damaged by vibratory earth motion, although the surface structures and access shafts are likely to be more vulnerable.

Isolation of a repository could be disrupted by tectonic activity and cause faulting, which may alter the hydrologic regime, or elevation and subsequent exposure through erosional processes. The tectonic stability of a host rock can be evaluated by investigating and delineating these tectonic processes of deformation and the rates of deformation. The processes and factors of the tectonic stability can be determined from the tectonic history and significant geologic structural features.

The occurrence of strong ground shaking at a repository site from local or regional earthquakes is not expected to have serious effects on the repository at depth (Dowding 1978), although some operational components of a waste isolation facility may be disrupted. The primary effect of earthquake occurrence is faulting, an important mode of tectonic deformation. Faulting may or may not be evident at the ground surface.

B.4 HYDROLOGIC CONSIDERATIONS

The hydrologic regime (the surface water and ground-water systems) at the repository site must be favorable to geologic isolation (Office of Waste Isolation 1977).

Surface hydrology includes the distribution and occurrence of water at the surface of the area. Large rivers and lakes represent collection areas for surface water from surrounding regions and may be areas where underground water is moving to the surface. Such areas will probably be avoided because of the risks of flooding and entrance of water into the repository workings.

Ground water is an important consideration in geologic site selection for two main reasons:

1. It is a valuable and widely used resource and a repository should not be located where it will affect the quality or availability to an unacceptable level.
2. Ground water is generally considered to be the most likely agent for transporting radioactivity away from the repository during its expected lifetime.

Ground water is present in varying degrees of saturation in nearly all subsurface earth materials. Also, all rock units have some permeability (although it may be small in some cases), and have hydraulic conductivity varying from relatively high to very low. Ground water can dissolve and transport radionuclides. Waste isolation requires that the properties of the host rock minimize transport of the waste and that the host rock be isolated from more permeable media. The ability of a disposal media to isolate radionuclides within a hydrologic regime is determined from the factors that govern hydrologic transport via the local and regional flow patterns.

The local flow regime of a repository site can be characterized by the geohydrologic properties of the host rock and of the hydraulic gradients (inducement to flow). Evaluation of the isolation potential of these components requires geologic studies, hydrologic testing, and analysis of water characteristics (de Marsily et al. 1977; ERDA 1976).

The geohydrologic character of the repository medium is concerned with intergranular fluid properties (Walton 1970). A rock substance is composed of minerals compacted and cemented or crystallized together into a matrix. Spaces between grains and cementation material (called pore space) can contain fluid. The percent of pore space in the total matrix is the porosity. The volume of fluid a repository medium can contain is described in terms of percent water saturation and porosity (secondary rock discontinuities also contribute to its fluid volume capacity). Pore space is an important property in determining: 1) the ability of a fluid to flow through a medium, 2) the volume of fluid flow and 3) the rate of flow. The evaluation of porosity for the repository medium includes the in-situ condition and the effect of radioactive waste-induced alteration, e.g., precipitation and/or solution. Porosity alone does not determine the permeability of a medium. For example, a shale has high porosity because of the clay size particles but is essentially impermeable because the pores are not interconnected or are so small that capillary forces dominate.

The flow potential of a repository medium depends on the interconnection of pore space (permeability) and the pressure differential. Geologic materials may be grouped in order of flow potential into aquifers and aquitards (confining units). The definition and delineation of these units require knowledge of the geologic stratigraphy and matrix and rock mass hydrologic properties of the medium. Conventional subsurface geologic techniques are used to define the lithologic and horizontal and vertical distribution of a flow unit. Fluid chemistry and core analysis of porosity and permeability allow the estimation of volume of fluid available for flow within a unit and of the hydrologic characteristics of the repository medium. Hydrologic field testing of the individual flow units completes the delineation of these units.

A flow unit may contain a large volume of fluid and a high permeability but require an inducement to flow (Davis and DeWiest 1966). A difference in hydraulic head (gradient) is necessary before a fluid will flow through a porous medium. A repository site contains local gradients, both vertical and horizontal, and a regional gradient for given flow potential units. A hydrologic gradient could exist because of elevation differences between the surface point where fluid enters the unit (recharge area) to the repository medium. Hydraulic gradient across the repository medium is generally determined by finding the difference in fluid level (potentiometric head) between wells of known depth.

The regional geohydrology is important to waste isolation in terms of conditions that may affect the local hydrologic regime. Possible effects include changes in hydraulic gradient from water usage or climatic changes.

Regional hydraulic parameters significant in maintaining isolation are recharge and discharge conditions. Recharge is of particular interest in establishing the volume of fluid available to an aquifer. Tectonic movements have the potential to significantly alter the hydraulic regime.

Hydrologic considerations enter into each stage of the site selection process. In the early stages the broad regional characteristics of surface and subsurface water flow are examined for compatibility with waste isolation. Regions may be eliminated from consideration on the basis of unfavorable characteristics, for example, high regional flow gradients, presence of aquifers near the proposed repository depth, or alteration of hydrologic regime from future climatic changes or tectonic events.

Other hydrologic characteristics may be of overriding importance to site selection. For example, interior drainage (surface runoff that does not drain to the ocean) is particularly well developed in the Great Basin of Nevada and Utah. The characteristics of such a hydrologic regime offer longer flow paths and greater travel times than does surface water flow to the migration of radionuclide wastes beyond the boundaries of the system. Other examples of favorable regional hydrologic conditions include arid climate and low hydraulic gradients (vertical and horizontal) in the surface and subsurface regimes.

Further consideration of hydrology involves more detailed characterization of the regional regimes that have passed the first phase. The collection of rock properties data

and the field measurement of hydrologic parameters enumerated will be needed. These data will be input to hydrologic models of candidate areas for locating sites at which geologic barriers are particularly effective.

Detailed evaluation of individual sites will be required in order to predict the complicated interaction of a repository model and hydrologic regimes. An important field of research, for instance, is the prediction of thermal effects on rock permeability near a repository.

Site selection will probably avoid areas of known major aquifers. In areas other than those with major aquifers, a preliminary ground-water characterization would certainly be a factor to be considered in the early stages.

To characterize an area's ground-water supply and potential requires determination of such factors as depth to producing zone(s), yield (usually determined from pumping tests) and an estimate of the supply available. From available wells, porosity, permeability and change in water level caused by pumping are measured. These aquifer properties and how they change with distance are extrapolated over the area if other measurements are not available. Usually other existing wells supply information to help determine the configuration of the water table or artesian pressure surface, direction of flow and estimates of rates of movement. These methods are generally applied to areas and rock units that yield water in usable quantities, whether on a scale for cities or for a single dwelling (Walton 1970, Davis and DeWiest 1966). They are not as applicable and have not been applied as widely to areas where porosity, permeability and yield are very low--that is, where usable supply cannot be obtained.

The areas of very low ground-water supply or flow are more favorable candidate areas for waste repositories because of the smaller opportunity for moving water to contact the waste and possibly transport it. Flow properties will need to be determined because of the time periods associated with a repository. Flow rates and velocities of ground water that are insignificant over a 50-yr period may be significant over hundreds to thousands of years. Methods of evaluating free water and its movement in media of these areas are available (for example, laboratory determinations of porosity and permeability are made from field core samples) but zones of fracture or joint flow are difficult to evaluate and describe in laboratory tests. Field tests will be necessary to measure in-situ properties after a potential site is chosen.

When possible, future climatic changes (for example, a change to a much wetter climate) should be considered with the attendant possible effects such as change in ground-water levels on a repository.

It seems reasonable to assume as one possibility that free water, over thousands of years, may enter the repository even in shale and possibly salt. The effect of the water will depend on the condition and state of the waste at the time, and transport of radionuclides will depend on the rate of water movement, if any, through the repository and the physical-chemical properties of the repository medium.

B.5 NATURAL RESOURCE CONSIDERATIONS

Known occurrence of any natural resource will make an area less suitable for a repository. Construction of a repository will effectively remove the resource from use or limit access to it, and will need to be weighed against economic value, need and supply of the resource. Care will be needed in estimating future need and predicting value of materials perhaps not considered to be resources today.

B.6 MULTIPLE GEOLOGIC BARRIERS

The multibarrier concept is a "defense-in-depth" or "multiple barrier" approach to offsetting the present lack of certainty or predictability in some factors of the waste disposal system. The basic purpose of the concept is to provide a series of independent barriers to radionuclide migration that taken together represent a compound or multiple barrier. The multibarrier concept includes basically two major elements: 1) the waste package, which consists of the waste in whatever form, any materials between the waste and its container, the container or canister and any overpack or material placed between the canister and the host rock; and 2) the material or naturally occurring barriers consisting of the geologic disposal medium, its dimensions, its properties, tectonic setting, properties of contiguous and surrounding rock materials and the disposal medium's position in the regional and local hydrologic systems.

Waste forms and canisters are discussed in Section 4.

The natural barriers consist first and most importantly of the repository host rock and its properties. The properties include its physical, chemical, thermal and hydrologic characteristics. The host rock with its properties provides the justification for geologic disposal and is the main element in containing the waste within the repository and in isolating the waste from man's environment over the long term. The disposal medium provides this isolation through the depth of burial within the medium below the land surface and by providing minimal or very low rate of movement pathways for transport.

For this Statement it is assumed that ground water is the most probable transporting agent over long time periods and the emphasis is thus on locating the repository in such a position and medium that it is as isolated as possible from ground water.

Four geologic media have been selected to illustrate the range of rock properties that need to be considered in a host rock for a radioactive waste repository. All four rock types possess properties that are favorable for waste isolation. These, as well as some unfavorable characteristics are discussed in the following pages.

B.6.1 Salt Deposit Properties

Salt (NaCl) deposits appropriate for disposal media occur in stratiform masses (bedded salts) and in salt domes. Salt deposits result from precipitation of halite (NaCl) by evaporation from seawater. Salt precipitation often alternates with the deposition of shale and carbonate minerals, resulting in salt deposits interbedded with other sedimentary rocks. Generally the degrees and mineralogical types of interbedding vary greatly. Salt domes are formed by the flow of bedded salts laterally to form masses which then move upward and deform and frequently penetrate overlying strata (diapirism). Salt flow is induced by the low specific gravity of salt plus variations in the lithostatic pressure and differential compaction of overlying sediments. Salt dome deposits are usually of higher purity, are more homogeneous and have fewer fluid inclusions than do bedded salts. Salt deposits

applicable as disposal media are situated in distinct sedimentary basins throughout many of the contiguous 48 states, as illustrated in Figure B.6.2 (Office of Waste Isolation 1978b).

The existence of salt beds and formations that are known to be hundreds of millions of years old testifies to their isolation from water and their stability. Salt deposit strength properties are relatively fair to good in the undisturbed state. Salt is basically isotropic with minimal cohesive strength. The result is a highly plastic medium that tends to move (creep) under earth pressures, increasing with greater depth and temperature. Creep tends to seal discontinuities but is difficult to stabilize in tunnel openings. Although heat tends to reduce strength, high thermal conductivity of salt is conducive to heat dissipation. A salt deposit may contain moisture in interbed materials and in small cavities as brine inclusions. These brine inclusions have been shown to migrate or move toward a heat source (ERDA 1976). Salt moisture, if present, leads to increased heat effects and to the potential for strength loss from solution action. Undisturbed salt beds are essentially impermeable (Office of Waste Isolation 1978a,b).

Rock types associated with salt deposits include anhydrite (CaSO_4), limestone (CaCO_3), dolomite ($\text{CaCO}_3 \text{ MgCO}_3$), and shale (SiO_2 , Al_2O_3 , Fe_2O_3 , FeO , MgO , CaO , Na_2O , K_2O). Halite is highly soluble (Office of Waste Isolation 1978a). More information is needed about ion exchange rate, reaction to radioactivity, and potential chemical reactions with salt deposits, related rock types, and waste materials.

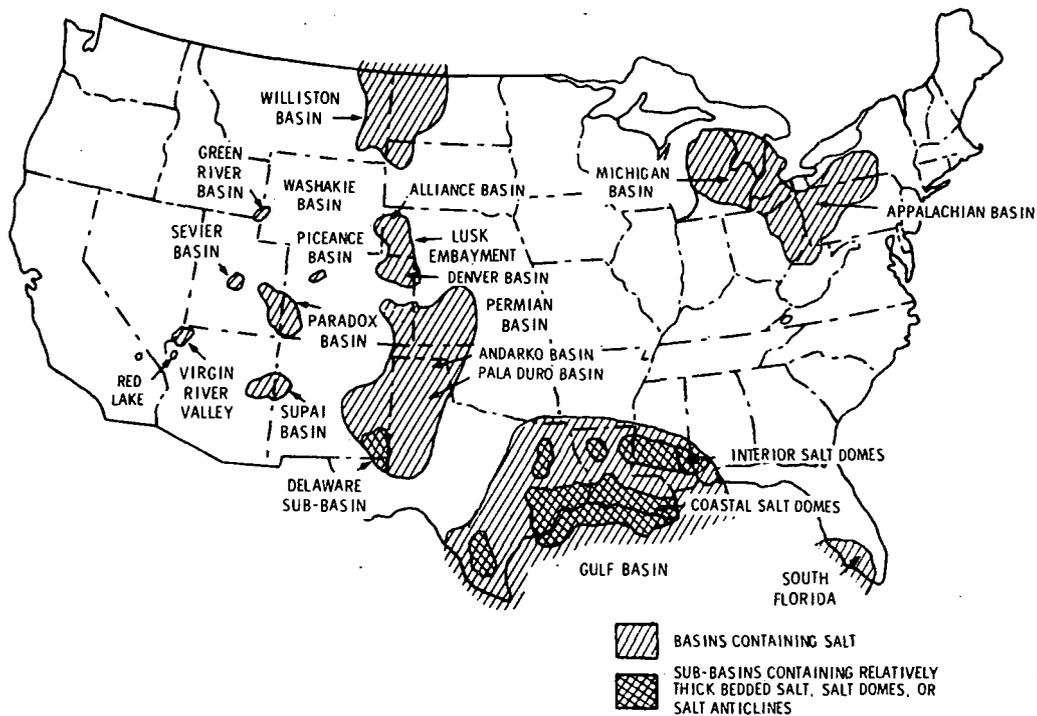


FIGURE B.6.1. Bedded Salt Deposits and Salt Domes in the United States (adapted from Office of Waste Isolation 1978a)

Salt deposit structures can be flat-lying, folded, or jointed. Jointing is generally parallel to bedding. Included within beds are large crystal masses, large rock masses of solidified impurities with lateral continuity, and lateral lithologic changes (Office of Waste Isolation 1978a). Joints can be anhydrite-filled, near vertical, unopen, moderately spaced, and generally extensive.

B.6.2 Granite Properties

Granite is an intrusive igneous rock with an equigranular, medium-to-coarse crystalline texture. It is generally light colored, composed principally of feldspar, quartz and, typically, hornblende and biotite. Granites are generally homogeneous in composition, with variations primarily in accessory minerals and secondary rock features. Granites are found as plutons, which are bodies of igneous rock that have formed beneath the earth's surface by consolidation from magma. Typical granite plutons include batholiths and smaller-scale stocks; they are very deeply rooted and enlarge with depth (Verhoogen et al. 1970, Holmes 1978).

Igneous rocks may have similar physical characteristics but range in chemical and mineralogical composition from granite to closely related rocks such as granodiorite. In many respects other closely related igneous rocks are similar to or identical to granite, but, because they vary significantly in major element, trace element and mineralogic composition, they are not considered to have the same disposal media properties as granite. The locations of potential repository granites within the contiguous 48 states are illustrated in Figure B.6.2. The areas identified represent large granite masses at or near the surface.

Granites are formed beneath the earth's surface. Their texture is a dense matrix of equigranular coarse grains. The porosity is low, with little or no natural moisture content. Intergranular permeability is extremely low. Also, strength is considered to be very high. Most component minerals are hard, resulting in high durability. Granites are generally very rigid, with little ability to deform under earth stress, but may exhibit fractures that could conduct water if they are open and water is available. Granites are basically resistant to temperature effects up to several hundred degrees Celsius. However, thermal expansion of particular minerals may be sufficient to cause fracture of the rock and possibly surface heave.

Granite is mostly composed of silica, alumina, and alkali elements, and forms minerals of quartz, feldspar, hornblende, and mica. Typical chemical composition of a granite is included in Ekren et al. (1974, Table 5.1.3). Mineral components of granite are almost inactive chemically under ambient temperature and pressure conditions. However, more data are needed about waste-granite reactions under repository conditions.

Granites have no bedding because of their intrusive igneous mode of formation, but may be layer-like. Joints tend to be blocky or sheet-like on a large scale, and their orientations may be vertical and intersect at right angles and/or horizontal and subparallel to the topographic surface. Joints, which range from sealed to partially opened and extensive often have little mineralization. Granite masses may contain dikes, veins and occasionally fragments of other rock material.

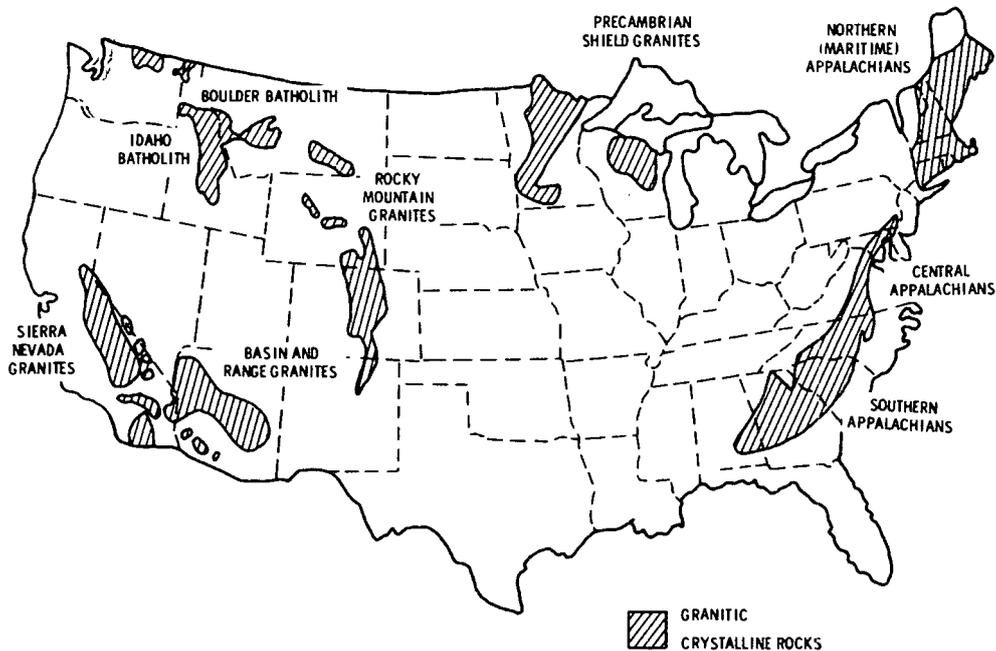


FIGURE B.6.2. Granitic Rock in the United States (adapted from Office of Waste Isolation 1978a)

B.6.3 Shale Properties

Shale is the product of the lithification or compaction and cementation of mud. Mud is predominantly composed of clay size particles (1/256 mm dia) and/or silt size particles (1/256 to 1/16 mm dia). The predominant constituents are clay minerals (hydrous aluminum silicates), and substantial amounts of mica, quartz, pyrite, and calcite (Table B.6.1) (Verhogen et al. 1970, Holmes 1978, Office of Waste Isolation 1978a). Mineral grains may either be poorly compacted in a soil-like manner or cemented like rock. Shales are in general stratified or laminated, and fissile, although some may show little layering and break into small angular blocks, as with mudstones. Shales are often interbedded with other sediments such as carbonates and sands. Shale units potentially applicable as disposal media are situated in sedimentary basins throughout many of the contiguous 48 states, as illustrated in Figure B.6.3 (Office of Waste Isolation 1978a).

Shales are relatively weak, partly because of the soft mineral components and weak cementation between grains. The general texture is fine-grained, and shale tends to split into flat, shell-like fragments in parallel bedding. The fine-grained clay minerals account for a very high natural moisture content and porosity. Because of fine pore size, intergranular permeability is low. Many shales have the ability to accommodate large deformations with a potential for plastic flow.

Clay minerals are known to have a high ion-exchange potential. Wetting and drying of shale will weaken the rock and may cause it to crumble. Shale may oxidize (as well as dry) when exposed to air, affecting both strength and volume characteristics. More data are

TABLE B.6.1. Average Chemical Composition by Oxides for Representative Disposal Media

Compound, % of Total	Rock Types		
	Granite	Shale	Basalt
SiO ₂	70.2	55.0	49.1
Al ₂ O ₃	14.5	21.0	15.7
Fe ₂ O ₃	1.6	5.0	5.4
FeO	1.8	1.5	6.4
CaO	1.9	1.6	9.0
Na ₂ O	3.4	0.8	3.1
K ₂ O	4.1	3.2	1.5
MgO	0.9	2.3	6.2
H ₂ O	0.8	8.1	1.6
X _y O	0.4	1.9	2.0

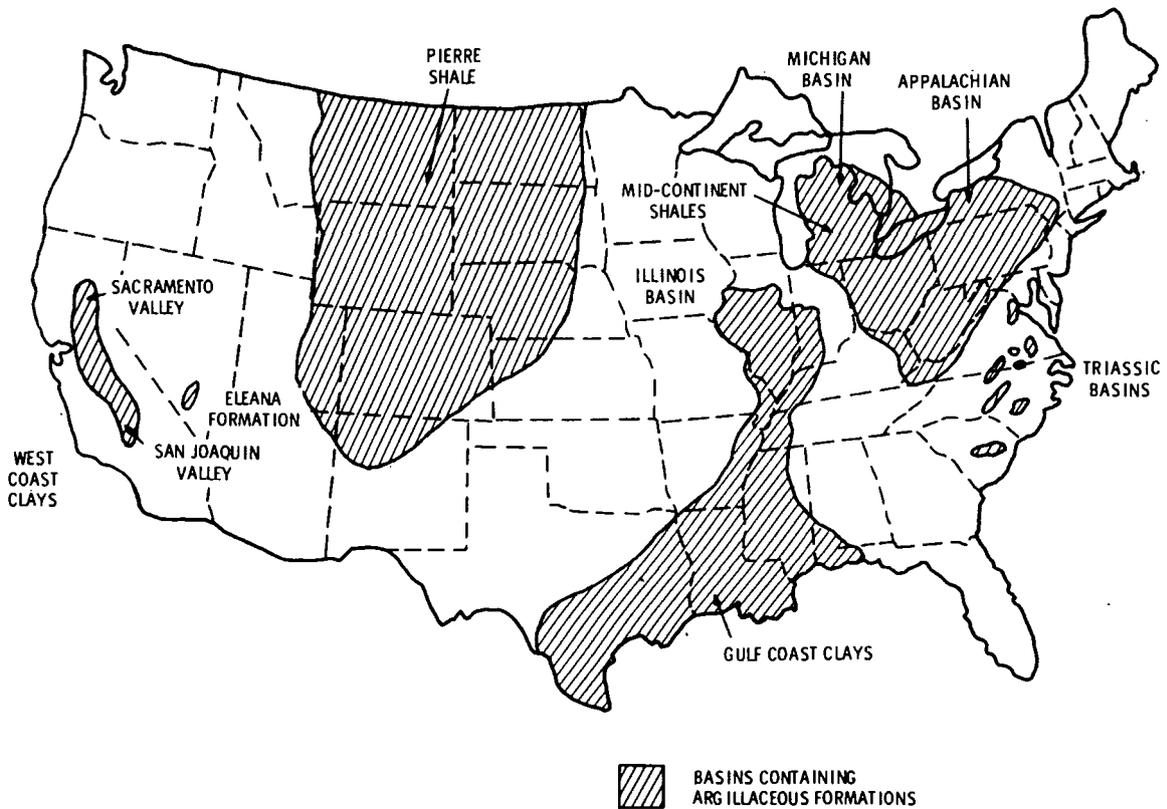


FIGURE B.6.3. Representative Shale Units in the United States (adapted from Office of Waste Isolation 1978a)

desirable regarding shale-waste reactions under repository conditions. Heating effects may be significant with shale as well as effects of temperature rise on contained water.

Shales may have discontinuities consisting of bedding, joints and fracture planes which are often filled with calcite, but also may be unfilled.

B.6.4 Basalt Properties

Terrestrial basalt flows are considered here to be applicable to conventional geologic disposal. Basalt is a black to medium gray, extrusive volcanic mafic rock (high in magnesium rock silicates) with the major mineral component calcic plagioclase (usually as phenocrysts) olivine and accessory minerals of magnetite, chlorite, sericite, and hematite (Office of Waste Isolation 1978e, Holmes 1978). The texture of a basalt may be either glassy or granular. Generally, basalt flows have a large areal extent. The locations of potential basalt repository areas are illustrated in Figure B.6.4. The basalts of southeastern Idaho are not considered because of high permeability features such as the Lost River and known large open lava tubes.

Basalt is commonly a very dense, high-strength material. Consequently, porosity and permeability are favorably low, with negligible moisture content, although interflow sedimentary units may be more permeable. Basalts remain relatively strong under elevated temperatures but may exhibit expansion. An average chemical composition of basalt is included Table B.6.2. More data are needed about basalt-waste reactions under repository conditions.

Joints are generally platy or columnar. They may be filled with various secondary minerals, alteration or weathering products of basalt. Joints may be unopened or opened with wide spacing (~ 0.3 - 1.8 m) and be smooth to rough. Joints in basalt may be extensive. They are generally unfavorable because of their potential for high permeability and ground water flow.

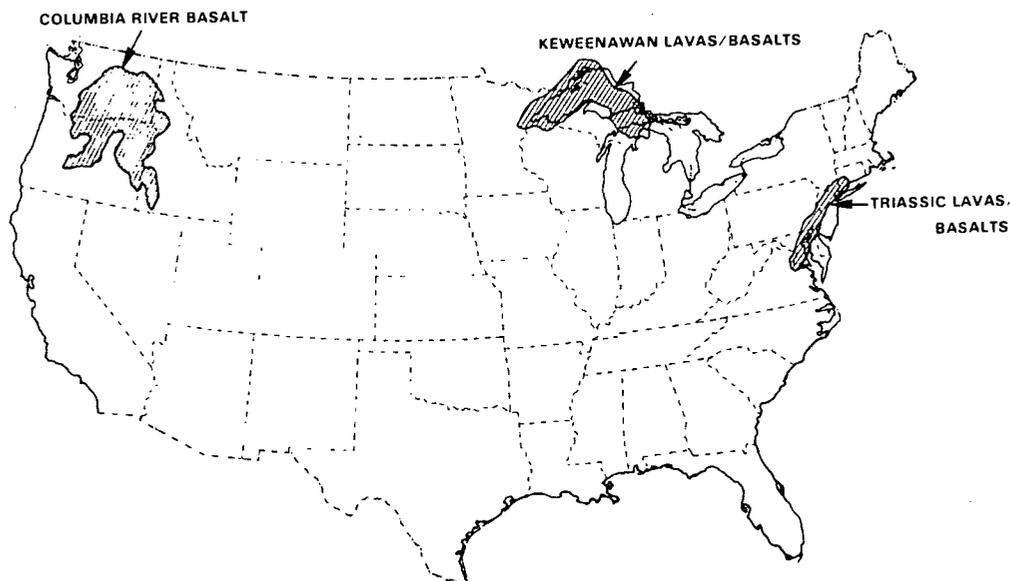


FIGURE B.6.4. Potential Repository Basalts in the United States
(adapted from Office of Waste Isolation 1978a, Dott and Batten 1971)

B.7 THE SITE SELECTION PROCESS

Locating a site for geologic disposal of nuclear wastes must necessarily proceed in a certain sequence to attain the best available combinations of conditions. This optimization of siting considerations is employed to offset the uncertainties of geologic prediction.

At each step, appropriate technical criteria as well as optional siting considerations are required to guide the work and facilitate judgments of suitability. Licensing criteria are under development by the Nuclear Regulatory Commission and performance criteria by the Department of Energy (Gray et al. 1976). Such criteria are based on the need to reduce to the maximum extent achievable the risk of radionuclides being released from the repository to the human environment.

The site-selection process can also take on a different character (Gray et al. 1976). Because the practical aspects of gaining access to land for reconnaissance and exploration, at least over the near term, may impose severe restrictions on the area considered (Gray et al. 1976), sites can be selected for detailed investigation based on ownership by appropriate government agencies. Although satisfaction of appropriate technical criteria and siting considerations is essential at each stage, other factors also are relevant to the site-selection process, and could dominate. Among these are ease and cost of access, distance from other societal activities, and societal acceptance of the locations as a candidate repository site. Thus, certain sections of the country may be considered unavailable for further siting even though preliminary reconnaissance indicates generally favorable geologic conditions.

Also, the criteria for suitability of a site cannot be specified in great detail because of the complexity of the geologic settings; it is possible that the selection of initial regions for investigation may be done partly on the basis of nontechnical factors. Whether the process is begun this way or by a strictly technical approach, sites will be examined in detail and compared against the underlying radiological and environmental safety criteria. In the discussion that follows, a sequence of purely technical and scientific decisions is assumed, although it is recognized that socioeconomic and institutional factors must be considered in the site-selection process.

A purely technical approach to site selection begins on a broad nationwide scale in Stage I. A few basic considerations are used to arrive at candidate regions. Candidate regions are evaluated on a finer scale in Stage II using other geologic considerations to arrive at candidate areas. Stage III consists of individual site evaluations leading to selection of an optimum site from among a small number of possible alternatives. This selection process provides a systematic method to narrow the geographic area to be studied from the nation as a whole to smaller identified regions to even smaller geographic areas and finally to a small number of alternate sites. At each step unsuitable areas are discarded.

Stage I of the selection process begins with tectonic and hydrologic considerations that can be applied on a broad national scale (see Figure B.7.1). For each consideration, criteria need to be defined to serve as a basis for eliminating unsuitable regions and

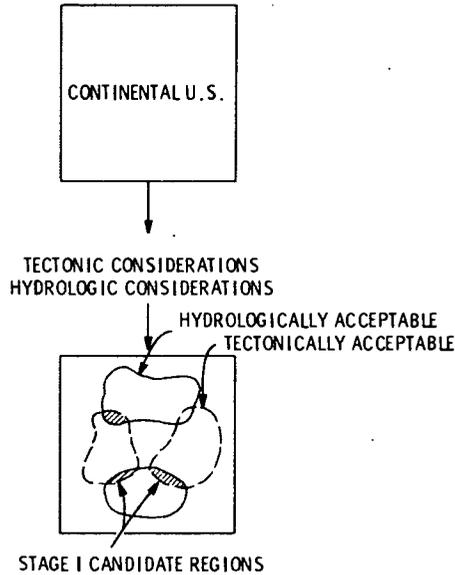


FIGURE B.7.1. Site Selection Process, Stage I

outlining the most suitable regions. Optimal choices for candidate regions are areas that satisfy both broad considerations. A hypothetical Stage I candidate region, for instance, could be an area that passes certain criteria both for optimal tectonic stability and hydrologic conditions. Selection of candidate regions can be accomplished by a thorough evaluation of available literature, existing geologic exploration data, and other existing information such as satellite imagery.

The candidate regions defined in Stage I enter into Stage II of the site selection process (see Figure B.7.2). General geologic considerations are applied on a scale appropriate to regional study, and criteria are again established to select areas with the most acceptable characteristics. A similar process is followed for each additional consideration (i.e., regional tectonics, hydrology, and depth). Optimal choices for candidate siting areas are those that have satisfied all Stage II considerations.

Data base additions required for evaluation in Stage II include extensive geologic mapping, generic research on rock properties (particularly their temperature dependence), characterizations of regional hydrology, climatic data, and instrumental data such as that obtained from geodetic, geophysical and microseismic networks.

A major task in Stage II will be to determine the activity or inactivity of fault systems within candidate areas. Repository siting will be ruled out within a designated distance from active faults to protect against possible fault rupture and the effects of strong seismic shaking. Repository siting criteria for seismic hazards have not been established. However, they may resemble current criteria of the Nuclear Regulatory Commission for siting of nuclear power plants.

The results of the above studies will be basic input to hydrologic considerations and Stage III modeling of specific sites. Stage III will require collection of as much

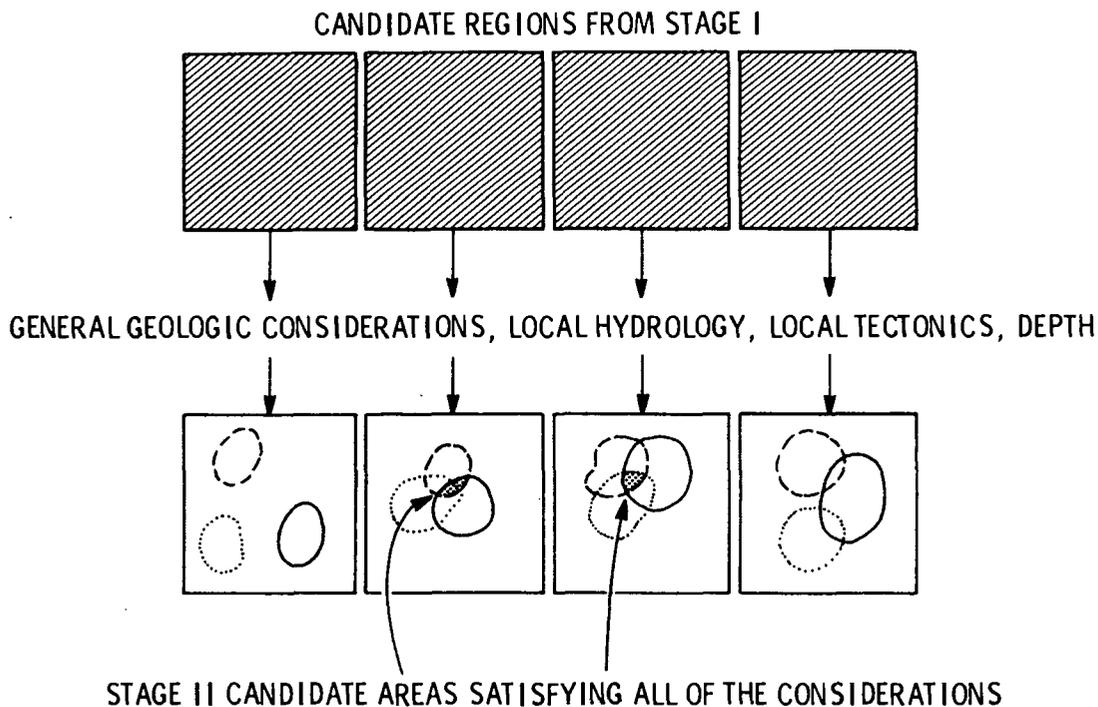


FIGURE B.7.2. Site Selection Process, Stage II

stratigraphic and structural data as possible without jeopardizing the isolation potential of the site. Drill holes, for example, are possible pathways for movement of water and loss of containment.

The candidate siting areas that result from Stage II enter into Stage III of the site selection process (see Figure B.7.3). All siting considerations are now applied on a site-specific scale. Again, under each consideration, criteria are used to eliminate unsuitable areas and to locate suitable sites.

Additional data base requirements (see Figure B.7.4) for Stage III are detailed site exploration data obtained by drilling, geophysical measurements, and possibly the opening of test tunnels. In-situ measurements of site-specific rock properties, state of stress, and hydrology will be conducted to the extent possible without compromising the future integrity of the repository.

It is possible that no site will be found to satisfy all criteria in Stage III. Trade-offs then may have to be made, which may reduce ideal conditions under one criterion, yet results in an acceptable site for better overall performance. An optimum site and alternatives are chosen and ranked in case unforeseen field conditions or sociopolitical factors prevent the use of one or more sites.

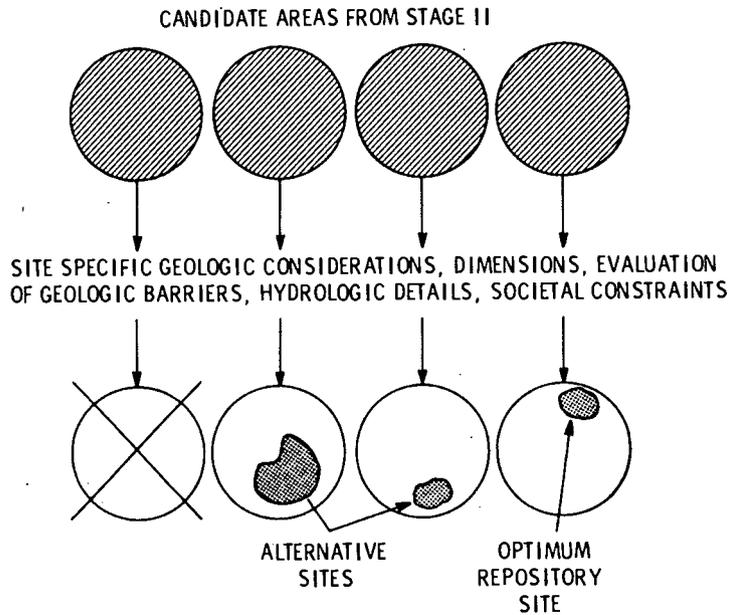


FIGURE B.7.3. Site-Selection Process, Stage III

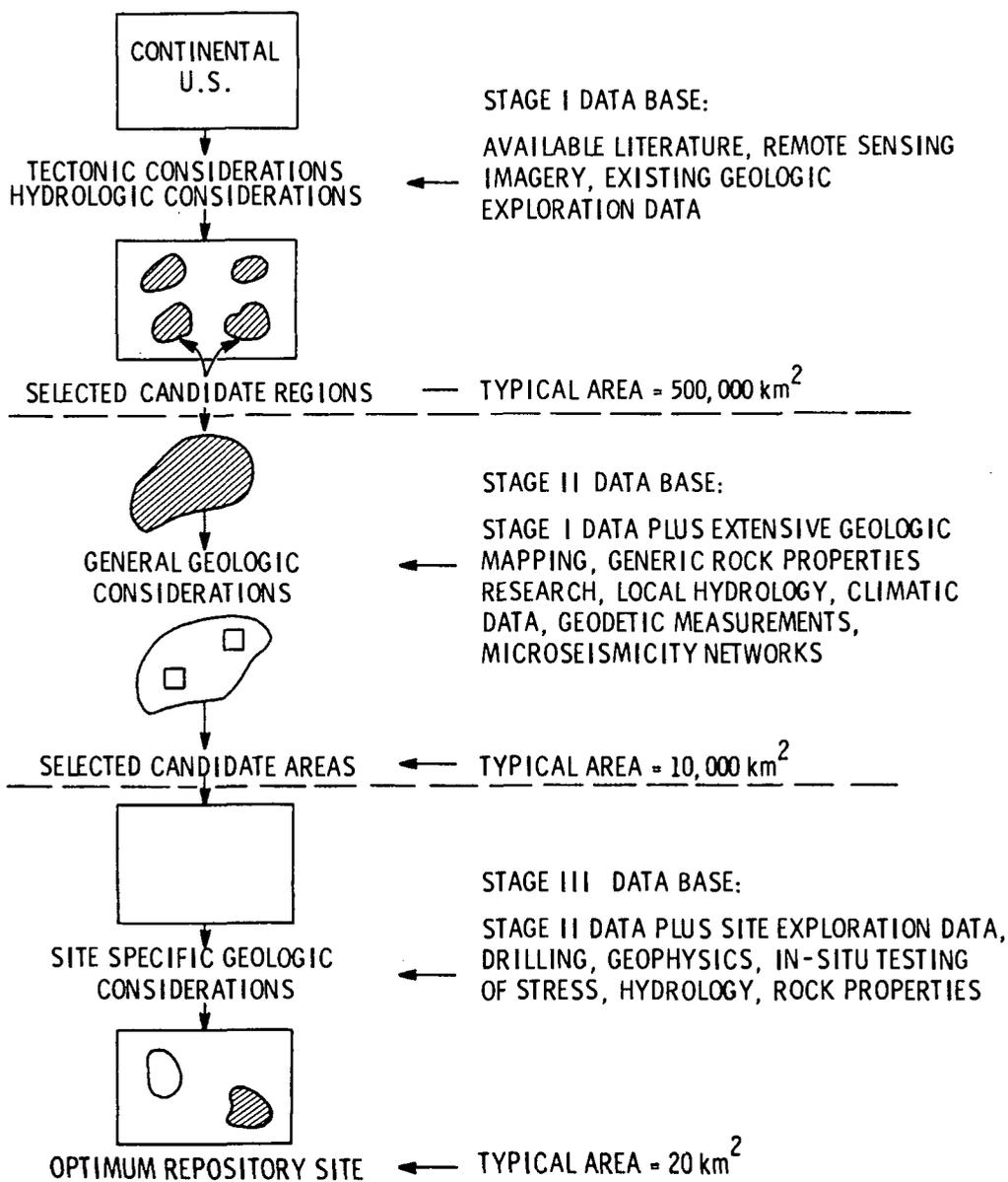


FIGURE B.7.4. Additional Data Base Requirements

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APPENDIX C

RADIOLOGICAL STANDARDS

Numerical values of annual dose limits have been set by the Nuclear Regulatory Commission (NRC) and Department of Energy (DOE). These limits and the Concentration Guides (10 CFR 20) derived from them are based on limits for occupationally exposed workers recommended by the National Committee on Radiation Protection and Measurements (NCRP 1957, 1959) and the International Commission on Radiological Protection (ICRP 1958, 1959). Minor modifications were made as a result of Federal Radiation Council (FRC) recommendations (1960) and more recent NCRP recommendations (1971). A review of the known biological effects of ionizing radiation by the National Academy of Sciences-National Research Council (NAS-NRC 1972) confirmed an earlier recommendation for limiting genetic exposure of the population, which corresponded to that of the NCRP. All these scientific bodies considered available data on both immediate and delayed effects:

- medical data on effects following therapeutic use of external radiation sources such as X-rays, and of radionuclides such as radium and iodine.
- occupational accident data on exposure of radiologists, X-ray and cyclotron workers, and workers in nuclear industry
- observations on population groups such as atomic bomb survivors and those irradiated by heavy nuclear weapons test fallout near the Marshall Islands.

Delayed effects, observable only years after exposure, were inferred from consideration of data from animal experimentation, from available epidemiological statistics, and from a limited number of case observations from medicine and industry (most notably a group of radium dial painters). The potential effects considered were 1) genetic effects and 2) somatic effects, including leukemia, skin changes, neoplasms, cataracts, changes in life span, and effects on growth and development. The delayed effects produced by ionizing radiation in an individual are not unique to radiation. For the most part they are indistinguishable from conditions normally present in the population, which may be induced by other causes.

In deriving the 10 CFR 20 Concentration Guides, a uniform exposure period of 50 years for adults was used. When dealing with intakes of radionuclides with effective half-lives in the body of less than 90 days, or where calculating doses directly from air and water concentrations by ratio to the appropriate Concentration Guides, the number of years of exposure makes little difference in the dose calculations. However, problems arise for non-uniform exposures to radionuclides with longer effective half-lives, especially when dealing with several exposure pathways and a heterogeneous population of varying ages and local residence periods. Although ICRP publications (1968, 1971) aid in making dose calculations, proper application of annual dose limits in such instances is controversial. The implied method is to calculate a total dose to an organ for a "standard man" for 50 years including

the year of intake, and charge this total dose to that year for comparison with the dose standard. Alternatively, the total radionuclide intake for the year is compared to the annual intake used in calculating the Concentration Guide, and the resulting ratio is used. The latter is in keeping with the latest ICRP guidance (1977). In all cases the internal dose should be added to any dose from external sources.

According to Taylor (1973), the basic recommendation of both the NCRP and ICRP was that individuals in an exposed population (without the medical supervision given the worker and with no direct benefit from such incidental exposure) should not receive in excess of 1/10 the maximum permissible dose of radiation workers. As an allowance for the variability of exposure and the variable susceptibility to radiation effects of the general population (which includes different age groups, genetic backgrounds, and both sexes), the Radiation Protection Guides (RPG) (dose limits) of the FRC (1960) were further reduced by a factor of three for the average of general population groups. The resulting RPG of 0.17 rem per year average whole body dose for population groups coincided with the later ICRP recommendations (1964) for limiting average gonad dose of the population, based on the possibility of genetic effects, to 5 rem in 30 years, excluding medical exposures.

From these studies, the present guidelines have been derived. The "as low as reasonably achievable" guide, and the limits derived from 10 CFR 20 and other analyses, are identified below.

C.1 "AS LOW AS REASONABLY ACHIEVABLE" APPLICATION

The degree of risk to people from very low radiation doses is not apt to be answered by actual observations, now or in the future, because of the indicated low probability (ERDA 1975) of any observable health effect in individuals and the nonspecific nature of some effects. Although the ICRP and NCRP have previously recognized as working hypotheses the presumably conservative assumptions that all radiation effects would be linear with dose, have a zero threshold, and be independent of dose rate, the NCRP has reiterated its stand (1975) against using these assumptions for deriving numerical values for risk-benefit calculations. More recently, the ICRP (1977) has attempted to distinguish between certain somatic effects for which a threshold dose seems applicable and other somatic (primarily neoplasms) and genetic effects for which the zero threshold, linear hypotheses still should be applied. In any case, the basic principle of radiation protection is still that all radiation exposures of people should be kept to the lowest levels technically and economically practicable.

The Nuclear Regulatory Commission's 10 CFR 50 Appendix I (1975) defines "as low as reasonably achievable" (ALARA) population dose limits for light-water-cooled nuclear reactor effluents, primarily for design guidance, but also as an action level for operational control. Other nuclear facilities are not specifically covered. The NRC in the published summary of its formal opinion has adopted the use of the phrase "as low as is reasonably achievable" (as recommended by the ICRP in 1973) as a substitute for "as low as practicable," because ALARA is a more precise definition of the intention of this regulation. The numerical values of limits assigned by the NRC, for design guidance for each light-water reactor, are that whole-body doses to any individual shall not exceed 3 mrem per year from liquid effluents or 5 mrem per year from external radiation resulting from gaseous effluents.

At present, the dose limits cited in Section 2.2.1 still prescribe upper boundaries for permissible doses to people. Some fractions of these limits (or the corresponding Concentration Guides) are generally understood to be "as low as reasonably achievable" for routine waste management operations. Whether those should be 0.1, 0.01, or some other fractions of the dose limits can be evaluated for each facility and effluent stream only on a case-by-case basis by considering the effluent treatments and controls available and the costs of providing such treatment or controls.

C.2 DERIVED LIMITS AND ACTION LEVELS

In common with other radiation standards recommendations (NCRP 1959; ICRP 1959, 1977), 10 CFR 20 provides equivalent or alternative criteria as well as basic standards. The relationships among the several kinds of radiation standards criteria may be more easily understood by reference to Table C.2.1. This table relates various standards and guides to the stages between a source of radioactivity and a potential end point (health effect). Also shown are parameters that must be quantified for calculation between one step and the next (in either direction), as well as the measurements required to provide a basis for comparison with the appropriate standards criteria. The Regulatory Guides issued by the NRC provide generally accepted values and procedures for such quantification. No current standards provide specific limits in terms of health effects, although other criteria may imply acceptance of some level of probability of health effects.

It has been common practice to use the Concentration Guides for air and water given in 10 CFR 20 for direct comparison with environmental measurements of these media. However, without additional data, use of the Concentration Guides alone may lead to neglect of a significant pathway of population exposure. This can occur not only because other pathways of exposure may contribute to dose, but also because reconcentration or bioaccumulation processes may affect concentrations in other sources of intake or exposure. Alternatively, summing of fractions of Concentration Guides for a mixture of radionuclides may result in an overestimate of dose if the several nuclides behave differently in the body.

TABLE C.2.1. Comparison Chart of Radiation Standards and Recommendations

Stage	Factors	Bases for Evaluation	Standards or Criteria
Inventory	Quantities, physical and chemical forms	Measurements of containers, shipping records	Inventory Limits
Release	Release fractions, rates of release, effluent concentrations	Measurements of effluent	Release Guides, Operating Limits
Dispersion and/or Reconcentration	Meteorology, biology, hydrology, physical and chemical forms, concentration factors	Measurements of environmental concentrations, calculations	Concentration Guides
Intake and Exposure	Exposure periods, consumption rates, retention factors	Measurements of direct radiation, calculations, bioassays, whole-body counting	Intake Ranges - FRC, Annual limits of Intake - ICRP
Dose	Biological half-lives, distributions in body, body dimensions, radiation types and energies	Dose calculations for maximum individual and population average	Dose limits - 10 CFR 20, 40 CFR 190-191, NCRP Reports, ICRP Reports
Health Effect	Dose/response relationships, demography	Calculated probabilities of specific effects	ICRP 26, 27, 28

Reliance on comparison of environmental concentration in air and water with the Concentration Guides requires additional caution, because the Guides are based on assumptions of standardized intake rates of air and water (20 m^3 of air and 1.2 liters of water per day for adults, with an additional intake of one liter of equivalent water at the same concentration in foods), as well as continuous exposure for periods of up to 50 years. Age-dependency of dose/intake ratios was not included in the derivations except for radioiodines in the infant thyroid. A result of the methodology is that an environmental concentration exceeding the Concentration Guide only briefly may scarcely affect the annual dose. Such an occurrence, however, would signal the need for investigation and possibly corrective action.

Although population doses can and should be calculated for comparison with the basic standards, the time lag and measurement sensitivities associated with most environmental measurements usually make it necessary to derive operating limits (or working limits) to be applied at the sources, i.e., the effluent streams.

Figure C.2.1 shows the generalized relationships between various levels of environmental concentrations (or effluent releases). The lowest level is the background measurement that would have been observed at the point of sampling if the operations under consideration did not exist. Some increases in concentrations may result from normal operations. An environmental impact (in the sense of a concentration difference) is the difference between an environmental level due only to background (which may include a contribution from other sources such as fallout) and the level due to background plus normal operations. Control of that impact is subject to the application of the ALARA principle. Both the "Normal Background" as well as the "Normal Background plus Normal Operations" are in reality, distributions (rather than point values) that may and often do overlap or coincide.

Concentration Guides and external dose limits provide upper limits on acceptable release rates of radionuclides to the environment. Derived working limits or action levels refer to in-plant actions by management, such as redirecting an effluent stream to a freshly regenerated radionuclide absorber, and not to emergency actions outside the plant

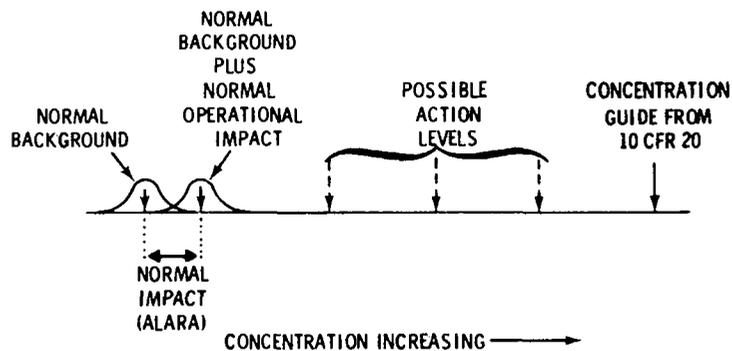


FIGURE C.2.1. Relationship of Operating Levels, Action Levels, and Concentration Guides (not to scale)

boundary (e.g., evacuation). These action levels are commonly set between the Concentration Guides and the levels due to background plus normal operations. Since a Concentration Guide is a definite value and the background value is a distribution which is largely site-determined, selection of not only an ALARA impact but also any "Action Levels" will depend upon cost-benefit-risk considerations. In practice there will normally be a series of graded action levels, with the lowest only an "investigation level." For example, as a design objective the allowed impact due to normal operation might be "set" at 1% of the Concentration Guide and an immediate remedial action level might be established at that point. A working limit, or investigation level, might in addition be set based on some multiple of the expected normal impact, provided that was still lower than the remedial action level.

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APPENDIX D

MODELS USED IN DOSE CALCULATIONS^(a)

Calculational models and parameters were used in evaluating the radiological dose to both regional and world populations. The regional dose calculations are discussed for chronic and accidental releases. The worldwide dose considers the distribution of tritium, carbon-14 and krypton-85.

D.1 DOSE TO REGIONAL POPULATION

The doses caused by chronic and accidental releases of gaseous and liquid effluents from the facilities and processes investigated in this study were estimated using several calculational models. The models and parameters used were selected to give a realistic but conservative appraisal.

D.1.1 Chronic ReleasesD.1.1.1 Air Concentration

The concentrations of radionuclides released in the atmosphere from these facilities were estimated using a Gaussian model (Slade 1968). Meteorological data on the joint frequency of occurrence of wind speed, wind direction, atmospheric stability and release parameters such as height and velocity for a particular plant were taken from the reference environment. The horizontal and vertical dispersion parameters, σ_y and σ_z , were taken from curves derived from the work of Pasquill and modified by Gifford (1977).

D.1.1.2 Air Submersion Dose

Air concentrations were estimated as outlined above for each of 16 sectors. For these sectors the centerline ground level dose was calculated for ten downwind distances from 1 to 80 km. Radiation doses to skin and to whole body were estimated from these air concentrations.

Both photons and beta particles can contribute significantly to the external dose to skin. The beta dose contribution is easily calculated using a semi-infinite cloud model. This model can be used because the range of beta particles in air is short compared to the dimensions of plumes considered. The gamma dose calculation is more complicated because of the relatively long range of photons in air. To properly determine the gamma contribution it is necessary to perform a space integration over the plume volume. The integration technique used in the reactor accident analysis computer program SUBDOSA (Streng et al. 1975 is

(a) In accordance with common practice, the term "dose," when applied to individuals and populations, is used in this report instead of the more precise term "dose equivalent" as defined by the International Commission on Radiation Units and Measurements (ICRU).

employed here except that the plume width is determined by sector boundaries rather than by a Gaussian concentration gradient. The contribution of gamma radiation to total-body dose was estimated by calculating the tissue dose at 5 cm depth. An occupancy factor may be used to account for the fraction of the year a person is exposed to the cloud. Also a shielding factor may be employed to correct for any shielding by buildings or structures between the recipient and the cloud.

D.1.1.3 Inhalation Dose

The air concentrations, derived as described above, were used along with the ventilation rate and dose factors to estimate the dose through the inhalation of radionuclides dispersed in the air.

The ventilation rate is the volume of air taken in by an individual per unit time. A value of 0.23 m^3/sec was used in this study (ICRP 1959).

The inhalation dose factor is given in units of rem/yr per Ci/yr intake and is dependent on the complex transport, retention, and elimination of radionuclides through the respiratory and gastrointestinal tracts. The model of the respiratory tract adopted by the Task Group on Lung Dynamics forms the general basis for the calculation of this dose factor (ICRP 1966). The computer code used for the calculations was DACRIN (Houston et al. 1974).

D.1.1.4 Ground Contamination Dose

Radionuclides from the air may settle on the ground, where they can accumulate during the time of the release. These can be a source of radiation for an individual or population groups.

This dose is determined using the 1) air concentration, 2) deposition "velocity" of the radionuclides traveling to the surface from the air, 3) an exponential expression which accounts for the accumulation of the radionuclide on the ground over a certain time period, 4) a dose factor, and 5) an occupancy factor.

The deposition "velocity" given in terms of m/sec is highly dependent on surface roughness, wind speed, and particle size. Based on many experimental studies, values of 0.001 m/sec for particles and 0.01 m/sec for iodine gas were selected for use in this report (Slade 1969).

The time over which the radionuclides accumulate in the soil is dependent on the lifetime of the facility releasing the material. In this study a value of 30 years is used, which is considered to be about the average lifetime of a nuclear facility.

The dose factor for the dose from ground irradiation is calculated by assuming that a receptor is 1 m above a large, nearly uniform, thin sheet of contamination (Soldat 1971, Fletcher and Dotson 1971). A factor of 0.5 to account for dose reduction due to ground surface roughness is also included in dose factors. These dose factors have units of rem/hr per pCi/m^2 of surface.

D.1.1.5 Ingestion of Food Crops

Food crops may become contaminated by deposition of radionuclides directly from the air or from irrigation water upon the plant surfaces or by radionuclides taken up from soil previously contaminated via air or water. Many factors must be considered when calculating doses via ingestion of these foods. These factors account for the movement of radionuclides from release to the receptor and form a complex sequence (Baker et al. 1976).

Equations used to calculate such doses are given in two parts: the first accounts for direct deposition onto leaves and translocation to the edible parts of the plant, while the second accounts for long-term accumulation in the soil and root uptake.

For sprinkler irrigation and for deposition of airborne materials both parts of the equation are used, while only the part dealing with root uptake is required for ditch irrigation. Tables of transfer factors and plant uptake factors are stored in files in the program FOOD (Baker 1977). The program can handle nine crops and their pathways to man. The output of the program lists the concentrations of radionuclides in the food crops and the fraction of the concentration due to each part of the equation (i.e., leaf or root). It also lists the dose to each organ from each nuclide/crop combination, with a summary of total doses from all crops and nuclides combined.

The nuclides ^3H and ^{14}C are treated as special cases in the FOOD program. The concentrations in the initial environmental media (air or water) are calculated on the basis of the specific activity of the nuclide in the naturally occurring stable element.

D.1.1.6 Ingestion of Animal Products

Five products--milk, eggs, beef, pork, poultry--are included in the FOOD program. The concentrations in the animals' feed are first calculated as discussed above for human food crops.

The equation, the quantities of animal feed and water consumed, and a listing of the transfer factors (fraction of each day's intake appearing per liter of milk or kilogram of eggs or meat) are given by Baker et al. (1976). The output of FOOD lists doses to various organs by nuclide and food type and summarizes total dose from all nuclides in milk, eggs, and meat (beef, pork and poultry).

D.1.1.7 Accumulated Doses from Foods

The computer program PABLM was written to calculate cumulative radiation dose to people from the ingestion of food. A total of eight food categories (leafy vegetables, other above-ground vegetables, root vegetables, fruit, grain, eggs, milk, and meat) can be selected with corresponding consumption rates, growing periods, and irrigation rates or atmospheric dilution parameters assigned by the user. Radionuclides may be deposited by water used for irrigation or directly from the atmosphere onto vegetation or the ground for the expected operating life of the facility. Dose commitments to the whole body and six internal organs from 186 radionuclides can be accumulated for a specified dose period. However, computer core space limitations restrict input considerations to only four organs

and 75 radionuclides. A summary of cumulative dose and percent contribution by nuclide for each food type is calculated. Radionuclide concentrations in soil, plants, and animal products are also calculated.

D.1.2 Accidental Releases

The dose to individuals exposed to a passing cloud of accidentally released radionuclides consists of external and internal components. The external radiation doses are calculated using the computer code SUBDOSA (1975), and the spatial distribution determined by the methods described in Meteorology and Atomic Energy (Slade 1968) and code XOQDOQ (Sagendorff and Goll 1977) for a semi-infinite cloud. External exposure results from both gamma radiation and beta particles emitted from radionuclides while they are airborne and external to the human receptor. This dose is dependent not only upon the type of radiation (i.e., gamma or beta) but also upon the energy of the radiation and the spatial distribution of the airborne radionuclides with respect to the receptor. The type and energy of radiation are characteristic of each radionuclide.

Because the range of beta particles in the air is only a few meters, the air concentration at ground level is sufficient to calculate the doses resulting from beta-emitting radionuclides. Ground-level air concentrations are not sufficient, however, for calculating the dose from gamma radiation. This is due to the relatively large range of gamma radiation in air. This range varies according to gamma energy and can be as long as a few hundred meters. As a result, the dose from external exposure to gamma radiation during cloud passage depends upon the air concentration at distances up to a few hundred meters. Thus the height of release has much less effect on gamma dose than it does on beta dose, particularly at close distances. As before for air submersion doses, both beta and gamma radiations contribute to skin dose; but only gamma radiation contributes to total-body dose (calculated at 5 cm depth).

Inhalation doses are calculated using the same models and codes used for chronic release except for increased ventilation rate (0.35 ℓ /sec) (Sagendorff and Goll 1977).

D.1.3 Dose to Biota Other Than Man

The doses to terrestrial and aquatic animals living within the influence of the nuclear facilities described in this report were not calculated separately. Two recent comprehensive reports (NAS-NRC 1971 and Garner 1972) have been concerned with radioactivity in the environment and pathways to biota other than man. Depending on the pathway being considered, terrestrial and aquatic organisms will receive either about the same radiation doses as man or somewhat greater doses. Although no guidelines have been established to set acceptable limits for radiation exposure to species other than man, it is generally agreed that the limits established for humans are also conservative for these species (Auerbach 1971).

The literature relating to radiation effects on organisms is extensive, but very few studies have been conducted on the effects of continuous low-level exposure to radiation

from ingested radionuclides on natural aquatic or terrestrial populations. The most recent and pertinent studies point out that, while the existence of extremely radiosensitive biota is possible and while increased radiosensitivity in organisms may result from environmental interactions, no biota have yet been discovered that show a sensitivity to radiation exposures as low as those anticipated in the area surrounding fuel cycle plants. The BEIR Report (NAS-NRC 1972) states in summary that evidence to date indicates that no other living organisms are very much more radiosensitive than man. Therefore, no detectable radiological impact is expected on the aquatic biota or terrestrial mammals as a result of the quantity of radionuclides to be released into the River R and into the air by fuel cycle plants.

D.1.4 Direct Radiation from Transportation

The method used to calculate the dose to persons along the shipping route from a vehicle containing radioactive material follows that developed in WASH-1238 (USAEC 1972).

The equation used to estimate population doses incorporates several factors that integrate the dose to an individual as the radiation source passes his location. The formula then integrates the dose to all persons within a designated population distribution. The factors considered are radiation source strength, velocity of the transport vehicle, population density in areas of exposure to passing source, attenuation factors due to *gamma* interactions with air, and buildup factor to account for the contribution of scattered radiation.

The Department of Transportation's regulations limit the radiation level allowable outside the transport container rather than restrict the container's contents. However, there is still a radioactivity content limit for each kind of packaging and for each toxicity grouping of radionuclides. Consequently, the shipping containers are designed and loaded with that regulatory limit in mind. For this calculation, based on the regulatory limit of 10 mrem/hr at 6 ft from the surface of the vehicle, the maximum radiation dose rate at 10 ft from the apparent center of the source was estimated to be 10 mrem/hr (USAEC 1972). The radioactive shipment on the vehicle was considered to be a point source for distances from the source of 100 ft or more.

The length of time an individual spends near a source is a determining factor in the total dose received; thus the velocity of the source is important. It was assumed that a long-haul, maximum-weight motor carrier shipment averages 720 miles per day and that a car-load rail shipment averages 200 miles per day. Based on a uniform distance traveled each day and uniform distribution of persons along the route, the cumulative radiation dose to the population is the same whether the vehicle is always moving at a constant rate of speed or is standing still part of the day. (Movement or lack of movement of the vehicle obviously will have an effect on the dose distribution among individuals within the exposed population.)

It was assumed that the average population density is 330 persons per square mile in the United States east of the Mississippi River and in California, and 110 persons per square mile in the other midwestern and western states. It is further assumed that no people live within 100 ft of the railroad or highway right-of-way. The dose to persons

farther than 2600 ft is negligible. The population was assumed to be uniformly distributed between 100 and 2600 ft on each side of the route, grouped at 100 ft intervals. Since the nuclear power facilities under consideration are assumed to have useful lifetimes of 30 years, the 70-year cumulative dose from transportation of wastes from a given facility is approximated by multiplying the annual dose by 30.

D.2 DOSE TO WORLDWIDE POPULATION

Worldwide population doses were calculated for the three radionuclides that are considered to be the major contributors to total-body dose rates and long-term dose commitments: ^3H , ^{14}C , and ^{85}Kr . A constant world population of 6.4×10^9 persons was used for this analysis. This value, which is based on a United Nations projection, was reported by Killough (1977) for the year 2000. It agrees with the value of 6.3×10^9 derived from the method of the Environmental Protection Agency (EPA) (1973) using projections based on a 1970 population of 3.56×10^9 persons and an annual growth rate of 1.9%.

A different method was used to determine the quantity of each of the radionuclides to which the population was exposed. For ^3H , dispersion was calculated using a seven-compartment model that considered diffusion into and out of latitudinal bands. The exposure of the population was calculated using assumed diets whose concentrations of ^3H were related to those in local surface waters. A specific activity approach was used for ^{14}C in which the concentration of ^{14}C per gram of carbon in people was assumed to be equal to that in atmospheric carbon dioxide. It was assumed that ^{85}Kr diffused readily across latitudinal bands so that in a few years the concentration was uniform throughout the world's atmosphere. The dosimetry for ^{85}Kr is based on external exposure of the body to a semi-infinite cloud containing this radionuclide, with no accumulation within the body or in any environmental reservoirs other than the air.

Although the method for each radionuclide is different, each probably estimates the population dose to within an order of magnitude. Additional uncertainty is therefore introduced when doses from all three radionuclides are totaled. Moreover, care must be exercised in comparing the relative contributions of these three radionuclides because of the different methods and because of the uncertainty inherent in each.

Each of the three methods is discussed below.

D.2.1 Tritium

Tritium (^3H) and tritium oxide released to the environment mix rapidly with the ambient water and become part of the hydrologic cycle. Tritium rains out or is washed out of the atmosphere almost entirely in the hemisphere in which it is released. Transport across latitudinal bands even in the same hemisphere is slow (Renné et al. 1975). As a result, the tritium released from facilities in the United States will reach peak environmental concentrations in the 30° to 50° latitude band of the northern hemisphere, where most of the world's population resides.

Baker (1976) has calculated the radiation doses received by local (50-mile radius), regional (eastern United States), and worldwide populations from a continuous release of 1 Ci/yr of ^3H to the atmosphere using the "box" model of Renné et al. (1975). The facility releasing the ^3H was assumed to be located in the Midwest. Although the magnitude of the dose to the local population is sensitive to the specific site chosen, the regional population dose should be similar for most midwestern sites. In addition, the world population dose depends upon the latitude band and not the longitude of the release point.

Baker's analysis indicated that for a constant world population of 3.8×10^9 persons, the collective population dose rate, at equilibrium with a continuous release of 1 Ci/yr of ^3H , was 1×10^{-2} man-rem/yr for all three population groups combined. Less than 10% of this dose was received by persons residing within 80 km of the plant site but about half was received by the eastern U.S. population during the initial pass of the ^3H released from the midwestern site. The actual dose to the regional U.S. population from a ^3H release to the atmosphere could range from near zero for plants situated on the eastern seaboard to values approximately equal in magnitude to the equilibrium worldwide population dose for plants situated in the West or Midwest.

In Baker's model (Baker and Soldat 1976) the ^3H content of water and food consumed by the world's population was assumed to be related to, but not necessarily as high as, the ^3H concentration in the surface waters of the appropriate latitude band. Even so, the population-weighted average surface water concentrations were higher than those obtained in the simpler model used by the EPA (1973 and 1974), which assumed mixing of the ^3H in the circulating ocean water of the northern hemisphere. As a result, Baker's calculations of dose to the world population (excluding the United States) are about seven times greater than those estimated by EPA.^(a)

For the commercial waste management study, the methods used by Baker were adopted with the exceptions of changing the world population from 3.8×10^9 persons to 6.4×10^9 persons and using a release time of 30 years in place of a continuous release out to equilibrium. The resulting dose factors per unit release are summarized in Table D.2.1.

TABLE D.2.1. Total-Body Dose Factors, and Dose Commitment Factors for the World Population (6.4×10^9 persons), man-rem per Ci/yr released^(a)

Radionuclide	Dose Factor		Accumulated Dose Factor (70/30) ^(d)	Dose Commitment Factor (70/1) ^(e)
	(1/1) ^(b)	(1/30) ^(c)		
^3H	4.7×10^{-4}	6.8×10^{-3}	2.4×10^{-1}	8.2×10^{-3}
^{14}C	2.4	7.2×10^1	4.0×10^3	1.7×10^2
^{85}Kr	3.1×10^{-5}	4.1×10^{-4}	1.4×10^{-2}	4.7×10^{-4}

- (a) Exclusive of contribution to eastern U.S. population dose from first passage of fuel reprocessing plant (FRP) gaseous effluents if FRP is located in the Midwest or West.
- (b) World population dose in first year after a 1-Ci release (instantaneous equilibrium).
- (c) Annual world population dose in the 30th year (year 2000) after 30 years of continuous release of 1 Ci/yr.
- (d) Seventy-year accumulated dose to the world population from 30 years of release at 1 Ci/yr followed by 40 years exposure to the residual environmental contamination.
- (e) Seventy-year dose commitment to the world population from a 1-year release of 1 Ci/yr to the environment plus continued exposure to the residual environmental contamination.

(a) The calculated U.S. population dose, however, is only two times higher for the Baker model than for the EPA Model. The net result is that the combined world population dose (including the U.S. population) is about three times higher via Baker's model than via the model used by EPA.

D.2.2 Carbon-14

Most ^{14}C released to the atmosphere from nuclear facilities will be in the form of carbon dioxide (CO_2), with possible traces of organic compounds released from certain specific processes within the nuclear fuel cycle. After mixing with the existing CO_2 in the atmosphere, the $^{14}\text{CO}_2$ can either become incorporated directly in plant material or washed out of the atmosphere onto land or water surfaces.

Most analyses of the long-term radiation doses to large population groups from ^{14}C include the following assumptions:

1. Carbon-14 is released to the atmosphere as CO_2 .
2. It mixes rapidly with all carbon in the world's atmosphere-- 6.2×10^{17} g (320 ppm CO_2).
3. Mechanisms that remove carbon into less accessible sinks such as the deep ocean or that dilute the $^{14}\text{CO}_2$ with increased CO_2 releases from future fossil-fuel combustion can be ignored.
4. The specific activity (that is, activity of ^{14}C per unit weight of carbon) in the tissues of man eventually equilibrates with that in the atmosphere.

More complicated models are possible. Machta (1973) developed a seven-compartment model for CO_2 , similar to the one discussed for ^3H . It was further modeled by the EPA (Magno et al. 1974 and Fowler et al. 1976) for use in predicting radiation doses to large populations from ^{14}C injected into the troposphere by the nuclear industry. The EPA model was used only to predict the specific activity of ^{14}C in the troposphere including, however, modifications for the sinks mentioned in assumption 3. Assumption 4 was then used to calculate dose to man. Fowler et al. (1976) included an estimate that 99% of man's ^{14}C intake is through food and only 1% is through inhalation.

Killough (1977) further modified the EPA seven-compartment model to incorporate newer data on diffusive vertical transport of CO_2 in the deep ocean and the relationship between the concentration of inorganic carbon in the ocean surface waters and the partial pressure of dissolved CO_2 . The computer code developed by Killough to implement the resulting model is documented in detail.

For purposes of the commercial waste management analysis, the conservative model outlined in assumptions 1 through 4 was adopted. This model was also adopted by the Nuclear Regulatory Commission (NRC) in its testimony at the Allied General Nuclear Services (AGNS) reprocessing plant license hearings (Eckerman 1974). By comparison the doses calculated using this simple approach are about 25% higher than those calculated by EPA (Fowler et al. 1976), 50% higher than those estimated by Baker (1976), and nearly seven times higher than those obtained by Killough (1977). The comparison with Killough is not, however, straightforward because of the assumptions of growing population and increasing CO_2 concentrations used by that author.

D.2.2.1 Dose Conversion Factors for Carbon-14

The assumptions that the specific activity of ^{14}C per gram of carbon in man eventually reaches equilibrium with that in the atmosphere and that there are 16.1 kg of carbon in the 70-kg body of Reference Man (ICRP 1959) lead to the derivation of dose and dose commitment factors as discussed in the following paragraphs.

At a release rate of 1 Ci/yr over 30 years the accumulated quantity of ^{14}C in the environment will be 30 Ci. At the end of an additional 40 years there will still be 30 Ci in the environment. Diluting 30 Ci in the 6.15×10^{17} g of carbon in the atmosphere (Killough 1977) yields a specific activity of

$$(30 \text{ Ci} \times 10^{12} \text{ pCi/Ci}) / (6.16 \times 10^{17} \text{ g}) = 4.87 \times 10^{-5} \text{ pCi/g}$$

The dose rate (DR) factor after 30 years of release can be calculated from the following equation (Soldat 1976):

$$\text{DR} = 0.0187 \text{ CE rem/yr}$$

where

$$\begin{aligned} C &= \text{concentration in body (pCi of } ^{14}\text{C per g of body tissue)} \\ &= (4.87 \times 10^{-5} \text{ pCi of } ^{14}\text{C per g of C}) (1.61 \times 10^4 \text{ g of C}) / (7 \times 10^4 \text{ g total body)} \\ &= 1.12 \times 10^{-5} \text{ pCi/g} \end{aligned}$$

$$E = 0.0538 \text{ (MeV/dis)} \cdot (\text{rem/rad}),$$

The factor 0.0187 is derived from the product of (0.037 dis/sec per pCi) (3.156×10^7 sec/yr) (1.602×10^{-8} g · rad/MeV).

Therefore

$$\begin{aligned} \text{DR} &= (0.0187) (1.12 \times 10^{-5}) (0.0538) \\ &= 1.13 \times 10^{-8} \text{ rem/yr per person} \end{aligned}$$

For 6.4×10^9 persons, the worldwide dose rate factor thus becomes 72.1 man-rem/yr after the release of 1 Ci/yr for 30 years.

The 70-year dose commitment (DC) factor, which is the sum of the dose during release and the dose after release has stopped, is calculated as follows:

$$\begin{aligned} \text{DC} &= \left[(0 + 72.1 \text{ man-rem/yr}) / 2 \right] (30 \text{ yr}) + \left[(72.1 \text{ man-rem/yr}) (40 \text{ yr}) \right] \\ &= 3970 \text{ man-rem per 1 Ci/yr released for 30 years.} \end{aligned}$$

These dose factors are summarized in Table D.2.1.

D.2.3 Krypton-85

When krypton-85 is released to the atmosphere it will mix rapidly with the atmosphere in the hemisphere in which it is released. After about 2 years it will also be fairly well mixed throughout the world's atmosphere. For purposes of this analysis, therefore, simple uniform worldwide mixing of ^{85}Kr in the world's atmosphere has been assumed. Similar assumptions have been used by the NRC in its testimony for the AGNS fuel reprocessing facility at Barnwell (Eckerman and Congel 1974) and the EPA in its projections of population dose commitments from the nuclear industry (EPA 1973 and 1974).

The National Council on radiation Protection and Measurements has published a discussion of the behavior and significance of ^{85}Kr in the atmosphere (NCRP 1975). In that report a comparison was made between the population exposure estimates made by detailed modeling of ^{85}Kr dispersion and estimates assuming uniform mixing in the world's atmosphere.

The model used in this analysis ignores the higher concentrations near the source and during the first pass through the latitudinal band where the release occurs. As a result, the model underestimates the local and regional dose at short times after the release. However, the net effect on the worldwide dose from long-term accumulated dose commitment exposure is small--about 10 to 20%, depending on whether the nuclear facility is sited in the Midwest or on the East Coast. The rapid mixing across the equator makes separate accounting of the northern and southern hemisphere population doses unnecessary.

D.2.3.1 Dose Conversion Factors for Krypton-85

The world's atmosphere contains $3.96 \times 10^{18} \text{ m}^3$ of air at standard temperature and pressure (NCRP 1975). The concentration of ^{85}Kr at any time is simply the cumulative amount released (corrected for radioactive decay) divided by the volume of the atmosphere. For a continuous uniform release rate of 1 Ci/yr, the concentration (C_t) of krypton becomes

$$C_t = \left[(1 \text{ Ci/yr}) (10^{12} \text{ pCi/Ci}) / (3.96 \times 10^{18} \text{ m}^3) \right] \left[1 - \exp(-\lambda t) \right] / \lambda$$

$$= (2.53 \times 10^{-7}) \left\{ \left[1 - \exp(-\lambda t) \right] / \lambda \right\} \text{ pCi/m}^3 \text{ per Ci/yr released}$$

where

λ = radiological decay constant for ^{85}Kr of 0.0648 per year
 t = years since start of release.

For 30 years of continuous release at 1 Ci/yr the expression $\left[1 - \exp(-\lambda t) \right] / \lambda$ becomes 13.2. This indicates that after 30 years 13.2 Ci remain in the environment out of the total of 30 Ci released. The concentration (C_{30}) then becomes

$$C_{30} = 2.53 \times 10^{-7} (13.2) = 3.34 \times 10^{-6} \text{ pCi/m}^3 \text{ per Ci/yr.}$$

The concentration during the next 40 years after the release has stopped is this 30th year concentration corrected for decay. Thus the total time-integrated concentration (TIC) is the sum of the combined expressions for concentration during the two time periods (0 to 30 years and 30 to 70 years). This yields the following equation:

$$\text{TIC} = (2.53 \times 10^{-7}) (1/\lambda^2) \left[\lambda t_1 + \exp(-\lambda t_2) - \exp(-\lambda \Delta t) \right] (\text{pCi}\cdot\text{yr}/\text{m}^3) \text{ per Ci/yr released}$$

where

t_1 = time over which release occurs,

t_2 = time over which dose is calculated,

$\Delta t = t_2 - t_1$.

For $t_1 = 30$ years and $t_2 = 70$ years the expression within brackets becomes 448, which yields a time-integrated concentration of

$$1.13 \times 10^{-4} \text{ pCi}\cdot\text{yr}/\text{m}^3 \text{ per Ci/yr.}$$

Unlike ^3H and ^{14}C , which emit only low-energy beta particles during their radioactive decay, ^{85}Kr emits a gamma photon in a small percentage of its decays. These photons plus a small contribution from bremsstrahlung associated with the beta decay are capable of irradiating the whole body^(a) during external exposure to ^{85}Kr dispersed in air. Krypton-85 is not significantly absorbed into the body during inhalation, and this pathway makes a negligible contribution to the whole-body dose (NCRP 1975).

Soldat et al. (1973) have calculated the whole-body dose factor for a person immersed in a half-infinite cloud of ^{85}Kr to be 2.2×10^{-3} mrem/hr per Ci/m^3 (1.9×10^{-8} rem/yr per pCi/m^3). Combining this dose factor and a constant world population of 6.4×10^9 persons with the expression for concentration (C_{30}) yields the world population whole-body dose rate in the 30th year as follows:

$$\begin{aligned} & \left[3.34 \times 10^{-6} (\text{pCi}/\text{m}^3) \text{ per } (\text{Ci}/\text{yr}) \right] (6.4 \times 10^9 \text{ persons}) \left[1.9 \times 10^{-8} (\text{rem}/\text{yr}) \text{ per } (\text{pCi}/\text{m}^3) \right] \\ & = 4.08 \times 10^{-4} \text{ man}\cdot\text{rem}/\text{yr} \text{ per Ci/yr released for 30 years.} \end{aligned}$$

The accumulated 70-year dose is

$$\begin{aligned} & \left[1.13 \times 10^{-4} (\text{pCi}\cdot\text{yr}/\text{m}^3) \text{ per } (\text{Ci}/\text{yr}) \right] (6.4 \times 10^9 \text{ persons}) \left[1.9 \times 10^{-8} (\text{rem}/\text{yr}) \text{ per } (\text{pCi}/\text{m}^3) \right] \\ & = 1.38 \times 10^{-2} \text{ man}\cdot\text{rem}/70 \text{ years per Ci/yr released for 30 years.} \end{aligned}$$

These factors are summarized in Table D.2.1.

(a) Defined as the layer of tissue lying 5 cm below the surface of the skin.

D.2.4 Dose Conversion Factors for System Analysis

The nuclear fuel cycle facilities in place and operating will change year by year. To obtain a realistic assessment of the long-term population dose commitments, calculation of the dose commitment from each year's operation followed by a summation of these yearly values is necessary. This can best be assessed by deriving population dose commitment factors for a one-year unit release.

Because of the nature of the three radionuclides involved in the world population dose estimates (^3H , ^{14}C , and ^{85}Kr), there is no long-term accumulation in the body. Hence, each year's release and resulting dose commitment can be treated independently of the others.

The following expression relates the 70-year dose commitment (from a 1-year chronic release) to the dose in the first year.

$$R = (1/\lambda^2) \left[\lambda t_1 + \exp(-\lambda t_2) - \exp(-\lambda \Delta t) \right] (\text{yr})^2$$

where

$$\begin{aligned} t_1 &= 1 \text{ year,} \\ t_2 &= 70 \text{ years,} \\ t &= t_2 - t_1 = 69 \text{ years,} \\ \lambda &= \text{radioactive decay constant (ln2/half-life).} \end{aligned}$$

The values of this ratio for ^3H , ^{14}C , and ^{85}Kr are given in Table D.2. Table D.2.2 also includes the dose commitment factors per unit release obtained when these ratios are applied to the first-year dose (item 1/1 from Table D.2.1).

Using these dose factors and annual releases of ^3H , ^{14}C , and ^{85}Kr from waste management facilities, estimates of worldwide population dose can be obtained for the evolving cycle systems.

TABLE D.2.2 70-Year World Population Dose Commitment from a 1-Year Chronic Release, man-rem/70 years per Ci/yr released

<u>Radionuclide</u>	<u>Ratio^(a)</u>	<u>Dose Commitment Factor^(b)</u>
^3H	17	8.2×10^{-3}
^{14}C	69	1.7×10^2
^{85}Kr	15	4.7×10^{-4}

(a) Ratio of 70-year dose commitment from a 1-year chronic release to the dose in the year of release.

(b) Seventy-year dose commitment to the world population from a 1-year release of 1 Ci to the environment plus continued exposure to the residual environmental contamination.

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APPENDIX E

RADIOLOGICALLY RELATED HEALTH EFFECTS

The radiation dose to man from ingestion, inhalation, or external exposure to specified quantities of radionuclides can be calculated with reasonable confidence. Estimates of the amounts of radioactive material that may be released from Commercial Waste Management (CWM) operations, however, and fractions reaching man via various environmental pathways are not as well defined. The relationship of dose to so-called "health effects" is even less well defined. Thus, estimates of "health effects" that may result from radiation exposure consequent to CWM activities can derive only from a chain of estimates of varying uncertainty. The usual practice in making these estimates is that if an error is to be made, it will be made in a way intended to overprotect the individual. As a result, if the chain of estimates is long, there may be considerable conservatism in the final value.

Because expected releases of radioactive materials are small, and the radiation dose to any individual is small, the effects considered are long-delayed somatic and genetic effects; these will occur, if at all, in a very small fraction of the persons exposed. Except as a consequence of the unusually severe accident involving larger doses, no possibility exists for an acute radiation effect. The effects that must be considered are 1) cancers that may result from whole body exposures, and more specifically, from radioactive materials deposited in lung, bone, and thyroid; and 2) genetic effects that are reflected in future generations because of exposure of the germ cells.

Knowledge of these delayed effects of low doses of radiation is necessarily indirect. This is because their incidence is too low to be observed against the much higher background incidence of similar effects from other causes. Thus, for example, it is not possible to attribute any specific number of human lung cancers to the plutonium present in everyone's lungs from weapons-test fallout, because lung cancers are known to be caused by other materials present in much more hazardous concentrations, and because lung cancers occurred before there was any plutonium. Even in controlled studies with experimental animals, one reaches a low incidence of effect that cannot be distinguished from the level of effect in unexposed animals, at exposure levels far higher than those predicted to result from CWM activities. Hence, one can only estimate a relationship between health effect and radiation dose, basing this estimate upon observations made at very much higher exposure levels, where effects have been observed in man, and carefully studied animal experiments. In this context the National Council on Radiation Protection and Measurements has said (NCRP 1975): "The NCRP wishes to caution governmental policy-making agencies of the unreasonableness of interpreting or assuming 'upper limit' estimates of carcinogenic risks at low radiation

levels derived by linear extrapolation from data obtained at high doses and dose rates, as actual risks, and of basing unduly restrictive policies on such interpretation or assumption" (NCRP 1975, p. 4).^(a)

An alternative approach involves direct comparison of the estimated radiation doses from CWM activities with the more accurately known radiation doses from other sources. This avoids the most uncertain step in estimating health effects (the dose-effect relationship) and provides a comparison with firmly established data on human exposure (i.e., the exposure to naturally occurring radiation and radioactive materials). Some people prefer to judge a risk's acceptability on knowledge that that risk is some certain fraction of an unquantifiable, but unavoidable, natural risk, than to base this judgement on an absolute estimate of future deaths that might be too high or too low by a large factor. Because of these judgmental problems it is the practice in this Statement to compare estimated radiation exposure from CWM activities with naturally occurring radiation exposure as well as to indicate estimates of cancer deaths and genetic effects.

E.1 LATE SOMATIC EFFECTS

Recently much literature has dealt with the prediction of late somatic effects of very low-level irradiation. This literature is not reviewed in detail here because it is recent and readily available. Instead, the various dose-effect relationships that have been proposed are briefly considered and justification is given for the range of values employed in this Statement.

Two publications have served as the basis for most recent efforts to quantify late somatic effects of irradiation. These are the so-called BEIR Report, issued in 1972^(b) by the National Academy of Sciences as a report of its Advisory Committee on the Biological Effects of Ionizing Radiations (NAS-NRC 1972); and the so-called UNSCEAR Report, a report to the General Assembly by the United Nations Scientific Committee on the Effects of Atomic Radiation, most recently revised in 1977 (UNSCEAR 1977).

Both the BEIR and UNSCEAR Reports draw their conclusions from human effects data derived from medical, occupational, accidental, or wartime exposures to a variety of radiation sources: external x-irradiation, atomic bomb gamma and neutron radiation, radium, radon and radon daughters, etc. These observations on humans were, of course, the result of exposures to relatively large total doses of radiation at relatively high dose rates. Their extrapolation to the low doses and dose rates of concern to us is acknowledged by the BEIR Report as "fraught with uncertainty" (p. 7). The BEIR Report concludes, however, that the assumption of a linear relationship between dose and effect, extending to zero dose with no threshold dose below which no effects are predicted, "in view of its more conservative implications,...

(a) EPA commented that this paragraph reflects a bias on the part of the authors. However, the NRCP quotation was chosen because it represented the negative point of view, and it was the purpose of this paragraph to reflect that point of view.

(b) A version of this report is in progress.

warrants use in determining public policy on radiation protection." But it further cautions that "explicit explanation and qualification of the assumptions and procedures involved in such risk estimates are called for to prevent their acceptance as scientific dogma" (p. 97).

The BEIR Report makes estimates of both absolute risk (cancer deaths per unit of radiation exposure) and relative risk (percentage increase above normal incidence of cancer deaths per unit of radiation exposure). And for each of these approaches it assumes either a 30-year or a duration-of-life interval following the latent period, during which risk remains elevated for non-leukemic cancer. Separate risk estimates are derived for the in utero, 0-9 years, and 10+ years age periods, reflecting presumed age differences in the sensitivity to radiation. The derivation of these risk estimates and their application to the U.S. population is summarized in the BEIR Report (p. 169) where the number of excess cancer deaths per year in the U.S. population, because of continual exposure at a rate of 0.1 rem/yr, is estimated as:

- 1726 for the absolute risk model with 30-year risk plateau
- 2001 for the absolute risk model with duration-of-life risk plateau
- 3174 for the relative risk model with 30-year risk plateau
- 9078 for the relative risk model with duration-of-life risk plateau.

The exposure rate of 0.1 rem/yr employed in these estimates is in the range of doses received from naturally occurring radiation sources in the continental U.S.

The BEIR Report risk estimates are shown in Table E.1.1, converted to a man-rem basis. This conversion involved dividing the risk estimates of Table 3-1, page 169, of the BEIR Report, by 20,000,000, since the U.S. population, taken as 200,000,000, if exposed to 0.1 rem/yr, receives a total annual exposure of 20,000,000 man-rem. The BEIR Report provides estimates for leukemia and for "all other cancers"; the "all other cancers" category is further subdivided for the absolute risk model as applied to those aged 10 or more. Values for bone and lung cancer are shown in Table E.1.1 as though the apportionment applied to the total population. It is important to note that the approximately five-fold range of values for total cancer deaths predicted by the four different BEIR Report models do not define a range between maximum and minimum possible values. They are merely four estimates, based on different assumptions, between which it is not possible to make a confident choice based on present knowledge.

The Environmental Protection Agency (EPA) in its Environmental Analyses of the Uranium Fuel Cycle (EPA 1973, 1976) chose single risk estimates, based on the BEIR Report, which it considered "the best available for the purpose of risk-cost benefit analyses, [while cautioning that] they cannot be used to accurately predict the number of casualties" (EPA 1973, p. C-14). These EPA risk estimates, expressed as cancer deaths per million man-rem, are also listed in Table E.1.1. The derivation of these numbers is not detailed in the EPA publications, but they continue to be used by the EPA and have been adopted by others.

TABLE E.1.1. Comparison of Various Estimates of Cancer Deaths per Million Man-Rem

Type of Cancer	BEIR Report (NAS-NRC 1972)				Environmental Protection Agency	Reactor Safety Study (NRC 1975)			UNSCEAR Report(2)	ICRP-26(10)
	Absolute Risk Model		Relative Risk Model			Upper Bound	Central Estimate(b)	Lower Bound(b)		
	30-Year Plateau	Life Plateau	30-Year Plateau	Life Plateau						
Leukemia	-26-(a)		-37-(a)		54 (1973a)	28	5.6	0	15-25	20
Non-leukemic	60	74	122	417		106	42	0		
Lung	16	19			60 (1973b)	22	4.4	0	25-50	20
Bone	2.4	3.0			16 (1973a)	7	1.4	0	2-5	5
Thyroid						13	2.6	0	5-15	5
Total	86	100	159	454	200 (1973b)	134	48	0	100	100

(a) 10-year risk plateau following in utero exposure, otherwise 25 years.

(b) Calculated on the assumption that no individual dose will exceed 10 rem.

NOTE: The term "health effects" is sometimes used to include sublethal cancers and less serious genetic defects. However, the estimates made in this statement are for cancer deaths and serious genetic effects.

The Reactor Safety Study^(a) (RSS) of the Nuclear Regulatory Commission (1975; this is commonly known as the Rasmussen Report) included an effort by an Advisory Group on Health effects to update and extend the conclusions of the BEIR Report (NRC 1975). Among the 17 members of this Advisory Group were five who also had served on the BEIR Committee. The RSS derived three classes of risk estimates: an "upper-bound estimate," a "central estimate," and a "lower-bound estimate." In contrast to the different BEIR Report risk estimates, the RSS estimates purport to establish a range within which the true value should be found. The RSS risk estimates for organs of interest to this Statement, and as applied to low-dose exposure, are listed in Table E.1.1. The details of the temporal exposure patterns, age distributions, and computational approaches employed in the BEIR and RSS Reports are not identical, and the risk estimates are therefore not strictly comparable; but errors from this source are negligible in comparison to the other uncertainties involved.

In arriving at upper-bound estimates, the RSS made two significant changes in BEIR assumptions and modified several numerical values on the basis of newer data. The "relative risk model" of the BEIR Report was eliminated and all estimates were based on the "absolute-risk model" and the plateau period for expression of non-leukemic cancer following postnatal exposure was taken as 30 years; the duration-of-life plateau option of the BEIR Report was dropped. The rationale for these changes is presented in the RSS Report. The major change resulting from new data was a 40% reduction in the leukemia risk of in-utero exposure; this was based upon revised dosimetry provided by the authors of the publication from which the BEIR risk estimate was primarily derived. The upper-bound estimates shown in Table E.1.1 are taken directly from Table VI 904, p. 9-33, of the RSS Report (NRC 1975), except for the thyroid cancer risk; this is derived from a "case" estimate of 134 per million man-rem modified by a mortality estimate of 10% (NRC 1975, p. 3-26 and 9-27).

The RSS central estimate "modifies the upper-bound estimate by correcting for risk reduction caused by both the ameliorating effects of dose protraction and the lesser effectiveness of very small acute doses" (NRC 1975, p. G-7). This correction acknowledges the preponderance of data from experimental studies, which indicate that the dose-effect relationship is not linear and that low doses of low LET (linear energy transfer) radiation delivered at low dose rates afford a significant opportunity for repair of radiation damage. The RSS discusses and references the extensive radiobiological literature on this subject and concludes that at doses below 10 rem, or at dose rates below 1 rem/day, a "dose-effectiveness factor" of 0.2 is justified (i.e., for a given total dose the dose effectiveness in producing a "health effect" is less at smaller dose rates). This was still considered a conservative position, the RSS Advisory Group on Health Effects was "of the unanimous opinion that the dose effectiveness factors they recommended probably overestimate the central estimate" (NRC 1975, p. 9-22). It should be recognized that some may not agree in applying such a factor in the human case, where the very limited data do not entirely

a) Since the Reactor Safety Study (RSS) represents the conclusions of a respected body of scientists, many of whom were also members of the BEIR Committee, the values reported in the RSS were not adopted but rather were considered when the values in Table E.1.2 were derived.

support the RSS position (Brown 1976). The EPA, in its formal review of the RSS study, disagreed with several aspects of the RSS health effects model, including the 0.2 dose rate effectiveness factor, and concluded that the RSS central estimate of late somatic effects "may be underestimated by a factor of 2 to 10" (EPA 1976).

Finally, the RSS acknowledges in its lower-bound estimate the possibility that a threshold for cancer induction may exist. While a threshold for primary radiation effects at the molecular level is considered unlikely on theoretical grounds, the mechanisms by which such effects become expressed as cancers are not understood, and available data in no way preclude the possibility of a threshold for these expressed effects. The RSS calculates its lower-bound estimate assuming a 10- or 25-rem threshold dose, either of which is larger than most doses predicted to occur to an individual from CWM activities.

The most recent and most thoroughly documented estimates of cancer risk from radiation exposure are those contained in the 1977 UNSCEAR Report. These values are listed in Table E.1.1. The UNSCEAR Report cautions that these values are ". . . derived essentially from mortalities induced at doses in excess of 100 rad. The value appropriate to the much lower dose levels involved in occupational exposure, and even more so in environmental exposures to radiation, may well be substantially less; . . ." (p. 414). Also shown in Table E.1.1 are the risk estimates adopted in the 1977 Recommendations of the International Commission on Radiological Protection (ICRP 1977), which were based primarily on the UNSCEAR Report.

E.2 GENETIC EFFECTS

It is known that genetic effects result from alterations within genes, called mutations, or from rearrangements of genes within chromosomes. There is no radiation-dose threshold for the production of mutations, but repair of damage to genetic material can occur during exposure at low dose rates. This information is reviewed and discussed at length in the 1977 UNSCEAR Report.

The conventional approach to this problem has been to estimate a "mutation doubling dose," i.e., the radiation dose required to double the existing mutation rate. The BEIR Report concludes that this doubling dose for humans lies in the range of 20 to 200 rem. The UNSCEAR Report considers additional experimental data and opts for a single value of 100 rem. Given a number for the doubling dose, if one can assume that radiation-induced mutations have the same effect on health as normally occurring mutations and if one knows the burden of human ill health attributable to such normally occurring mutations, one can directly estimate the genetic effect of any given radiation dose. Unfortunately, it is not clear that radiation-induced mutations are equivalent in effect to normally occurring mutations. Nor is there any confidently accepted quantification of the human ill health attributable to these normally occurring mutations.

Four kinds of specifically recognized genetically associated diseases are usually distinguished.

1. Autosomal dominant disorders are those caused by the presence of a single gene. The most common examples are: chondrodystrophy (abnormal cartilage development), osteogenesis imperfecta (abnormally brittle bones), neurofibromatosis (disease characterized by multiple soft tumors), eye anomalies including congenital cataract, and polydactylism (more than 10 fingers or toes) (Trimble and Doughty 1974). It is generally agreed that these disorders will double in frequency if the mutation rate is doubled (NAS-NRC 1972 and UNSCEAR 1977). There is some disagreement on their normal frequency of occurrence: the earlier data (Stevenson 1959) employed in the BEIR Report indicate a 1% normal incidence, while a more recent study of and Trimble and Doughty 1974), indicates an incidence of something less than 0.1%. These new data have not been fully accepted, however, and the 1977 UNSCEAR Report continues to employ the 1% normal incidence figures.
2. Multifactorial (irregularly inherited) disorders have a more complex and ill-defined pattern of inheritance. These diseases include a wide variety of congenital malformations and constitutional and degenerative diseases. Their normal incidence in the population was estimated in the BEIR Report to be about 4% (NAS-NRC 1972); however, the newer data of Doughty and Trimble suggest an incidence as high as 9-10% (UNSCEAR 1977). The BEIR Report states that, "The extent to which the incidence of these diseases depends on mutation is not known" but assumes a "mutational component" of 5 to 50% (p. 56). The 1977 UNSCEAR Report employs a single figure of 5% and considers 10% to be an upper limit (p. 429). Newcombe has argued that "the bulk of the most directly pertinent experimental studies thus

fail to demonstrate any important effect of irradiation on the irregularly inherited diseases, or on general health and well being," and concludes that "the collectively numerous irregularly inherited diseases of man are unlikely to be substantially increased in frequency by exposure of his germ plasm to radiation" (Newcombe 1975).

3. Disorders due to chromosomal aberrations include diseases characterized by changes in the number of chromosomes, or in the structural sequence within chromosomes. It is generally agreed that these diseases will show little increase as a result of low-level, low-LET irradiation, and they were not quantified in the BEIR Report. The 1977 UNSCEAR Report includes a numerical estimate for such effects.
4. Spontaneous abortions are known to occur as a result of chromosomal effects, often so early in pregnancy as to be undetectable. Such effects have been generally excluded as not a relevant health effect (NAS-NRC 1972).

In addition to the above specifically identifiable genetic effects, there may well be genetic influence on other unquantifiable aspects of physical and mental ill health. The BEIR Report assumed that two-tenths of this "ill health" was due to genetic factors related to mutation, acknowledging that "it may well be less, but few would argue that it is much higher" (p. 57). Using this factor and a mutation doubling dose of 100 rem, one calculates an eventual 0.2% increase in "ill-health" as a consequence of continual exposure to 1 rem per generation. Such ill-defined effects cannot be quantitatively compared to specific genetic effects, or carcinogenic effects, not can they be stated on a man-rem basis.

Table E.2.1 summarizes the BEIR Report and UNSCEAR Report genetic risk estimates. The EPA has employed an estimate of 300 genetic effects per million man-rem (EPA 1973, Part III), as has also the Medical Research Council in England (MRC 1975). The newer data on the normal frequency of autosomal dominant disorders (Trimble and Doughty 1974), and Newcombe's (1975) evaluation of the significance of multifactorial disorders, lead to an estimate for total genetic effects of only 10 per million man-rem. All of these estimates are for total effects, to be experienced over all future generations.

TABLE E.2.1. Estimates of Genetic Effects of Radiation Over All Generations

<u>Type of Effect</u>	<u>BEIR Report (1972)</u>	<u>UNSCEAR Report (1977)</u>	<u>EPA (1973)</u>	<u>Newcombe (1975)</u>
Autosomal Dominant Disorders	50-500	100		10
Chromosomal Disorders		40		
Multifactorial Disorders	10-1000	45		
Total	60-1500	185	300	10

E.3 CONCLUSIONS

For this Statement a range encompassing commonly used cancer risk factors has been employed, as indicated in Table E.3.1. At the same time the possibility of zero risk at very low exposure levels is not excluded by the available data. The lower range of risk estimates in Table E.3.1. may be considered more appropriate for comparison with the estimated risks of other energy technologies. The upper part of the range may be more appropriate for radiation protection considerations.

A range of 50 to 300 specific genetic effects to all generations per million man-rem was employed in this Statement. The lower value recommended by Newcombe has not been generally accepted and the upper end of the BEIR Report range seems too high in the light of newer evidence discussed in the 1977 UNSCEAR Report. As in the case of the somatic risk estimates, the lower end of the range may be considered more appropriate for comparative risk evaluations, while the upper end of the range may be appropriate to radiation protection considerations.

All estimates of health effects, as quoted elsewhere in this Statement, employ the risk factors summarized in Table E.3.1. No special risks are considered to be associated with any specific radionuclide except as reflected in the calculation of their dose equivalent (in rems) in the various tissues of concern. However, because of their particular significance, effects attributable to certain radionuclides (^3H , ^{14}C , ^{85}K , and plutonium) are discussed separately on the following pages.

TABLE E.3.1. Health Effects Risk Factors Employed
in This Statement

<u>Type of Risk</u>	<u>Predicted Incidence per 10⁶ man-rem</u>
Fatal cancers from:	
Total body exposure	50 to 500
Lung exposure	5 to 50
Bone exposure	2 to 10
Thyroid exposure	3 to 15
Specific genetic effects to all generations from total body exposure	<u>50 to 300</u>
Total	100 to 800

E.4 SPECIFIC CONSIDERATION OF HEALTH EFFECTS FROM TRANSURANICS

Data relevant for predicting specific health effects from transuranics have been considered elsewhere, in great detail (USAEC 1974, Bair 1974 and MRC 1975). Only the kinds of data available and the approaches that might be taken if specific transuranic health effect predictions were desired are considered here.

E.4.1 Experience with Transuranic Elements in Man

No serious health effects attributable to transuranic elements have been reported in man. There are extensive data, however, on exposure of man to transuranic elements. Such exposures arise from two main sources: the worldwide plutonium fallout from atmospheric testing of nuclear weapons and other devices, and the accidental exposure of persons working with transuranics. Since these exposures have produced no effects distinguishable from effects caused by other causes, the information is useful in health effects prediction only as an indication that unusual or unexpectedly severe effects are not to be anticipated; i.e., such negative data can be used only to set an upper limit on possible effects.

E.4.2 Experience with Natural Radiation in Man

Alpha-emitting elements are a natural part of the human environment. Humans have lived with these internally deposited radioelements and with radiation from other natural sources throughout the history of the species. It is of some relevance to note that inhaled naturally occurring alpha-emitting radionuclides contribute an average annual dose of about 0.1 rem to the lung, and that naturally occurring alpha emitters in bone contribute an average annual dose at bone surfaces of about 0.04 rem (NCRP 1975). While these doses cannot be related to any measure of specific effects, they have been at least "tolerable" on the evolutionary scale, and therefore slight increases can hardly have catastrophic effects.

E.4.3 Data from Experiments with Animals

Direct information on the toxicity of transuranic elements is available only from studies in experimental animals. The radiobiological literature suggests that the biological effects observed in such animal experiments will at least qualitatively approximate those that would occur in man exposed under the same conditions. Based on extensive data from several animal species, it is concluded that the most probable serious effects of long-term, low-level exposure to transuranics are lung, bone, and possibly liver tumors. Most of these data are from experiments with plutonium, but can probably be applied to other transuranics with less error than is involved in many other necessary assumptions. While quantitative extrapolation from animal to man involves considerable uncertainty, the animal data suggest tumor risks per million organ-rem of 60 to 200 for lung (Bair and Thomas 1976), and 10 to 100 for bone (Bair 1974, Mays et al. 1976). These estimates are compared with others in Table E.4.1.

TABLE E.4.1. Comparison of Transuranic Health Risk Estimates
(Tumor deaths per million organ-rem)

	BEIR (1972)		MRC (1975)	Mays et al. (1976)	Risk Estimates Based on Data from Animals
	High ^(a)	Low ^(b)			
Lung tumors	100	16	25		60-200 ^(c)
Bone tumors	17	2	5	4	10-100 ^(d)
Liver tumors			20		

(a) Relative risk model with lifetime plateau (Newcombe 1975).

(b) Absolute risk model with 30-year plateau (Necombe 1975).

(c) Data from Bair and Thomas (1976).

(d) Data from Bair (1974) and Mays et al. (1976).

E.4.4 Data on Effects of Other Types of Radiation on Man

Inferences concerning the effects of transuranic elements in man may be drawn from information available on the effects of other forms of ionizing radiation in man; e.g., data derived from medical, occupational, accidental, or wartime exposure of humans to different radiation sources, including external x-radiation, atomic bomb gamma and neutron radiation, radium, radon and short-lived radon decay products. Such information has been summarized in the BEIR and UNSCEAR Reports, as previously described. England's Medical Research Council (1975), considering much the same information covered in the BEIR and UNSCEAR Reports, derived risk estimates specifically applicable to plutonium.

Also of interest are recently accumulated data on the carcinogenicity of ^{224}Ra in human bone (Spiess and Mays 1971, 1972). These data are particularly relevant to risks from plutonium, since ^{224}Ra is predominantly an alpha emitter and, because of its very short half-life (3.64 days), irradiates only the surface layer of bone, in much the same manner as plutonium does. From these ^{224}Ra data, Mays et al. (1976) have estimated a bone cancer risk of 4 per million bone-rem.

Table E.4.1 compares tumor risk estimates from these several sources. Quantitative application of these data to the very low exposure levels involved in population exposure resulting from commercial waste management practices is uncertain; however, the kinds of data presented in Table E.4.1 are reassuring because of their general agreement, and because they predict no unusual incidence of effects not contemplated in the selection of the general risk estimates used in this Statement.

E.5 SPECIFIC CONSIDERATION OF HEALTH EFFECTS FROM KRYPTON-85

The radiological significance of ^{85}Kr was reviewed in a recent report of the National Council on Radiation Protection and Measurements (NCRP 1975). Most of the discussion in this appendix derives from that report, which should be consulted for details or for more extensive citation of the literature.

Because krypton is virtually inert chemically, it is not metabolized. Exposure of humans results from ^{85}Kr in the atmosphere external to the body, from ^{85}Kr inhaled into the lung, and to a much smaller degree from ^{85}Kr dissolved in body fluids and tissues. Over 99% of the decay energy of ^{85}Kr is in the form of a relatively weak beta ray (mean energy, 0.25 MEV) which limits the hazard from external exposure. There is general agreement that the dose to the sensitive cells of the skin from external exposure is about 100 times larger than the dose to the lung or any other internal organ (NCRP 1975, Kirk 1972, Soldat et al. 1975, Snyder et al. 1975).

The NCRP Report (1975) considers four categories of delayed effects from long-term exposure to low-level environmental concentrations of ^{85}Kr . These are: 1) genetic effects, 2) overall carcinogenic effects, 3) carcinogenic effects on skin, and 4) possible interaction of ionizing and ultraviolet radiation.

Estimation of genetic and overall carcinogenic effects of ^{85}Kr exposure involves no unusual features. Doses to gonads and to total body have been considered essentially identical by all who have considered the problem (NCRP 1975, Kirk 1972, Soldat et al. 1975). Genetic and carcinogenic risk factors chosen for general application in this Statement (Table E.1.2) should be appropriate to ^{85}Kr .

Carcinogenic effects on skin do constitute a unique problem, however, since the human exposure dose from ^{85}Kr is 100 times higher to the skin than to any other tissue. Dose-response data on radiation-induced skin cancer are limited, but suggest a threshold-type response; certainly the skin is less susceptible to radiation carcinogenesis than are many other tissues. The BEIR Report (Weston 1973), after review of the available data, concludes that "numerical estimates of risk at low dose levels would not seem to be warranted."

As a consequence, neither dose to skin nor estimated health effects that might result from low-level skin irradiation are presented in this Statement. (Skin cancer is perhaps the most easily controlled of all malignancies and is rarely fatal.)

The possibility of interaction between the radiation from ^{85}Kr and solar ultraviolet radiation, the latter of which is considered to be responsible for most human skin cancer, was raised in the NCRP Report (NCRP 1975). There is no direct evidence for such interaction, but the possibility was thought to justify further epidemiological and laboratory studies.

E.6 SPECIFIC CONSIDERATION OF HEALTH EFFECTS FROM TRITIUM

Although tritium is subject to the uncertainties involved in any prediction of effects at dose levels far below those for which there are experimental data, the relatively uniform distribution of hydrogen throughout the body and our understanding of the metabolism of hydrogen and water by the body do provide more confident dosimetry than is available for most other radionuclides. If there is special concern about tritium effects, it relates primarily to the difficulties of preventing its release to the environment, and to its worldwide distribution and availability to man following release. Many aspects of the biological concerns for tritium in the biosphere are reviewed in the Proceedings of a symposium on the subject, held in 1972 (Moghissi and Carter 1973).

There has been some concern that tritium incorporated in organic compounds, either before or following ingestion by man, might present a substantially increased hazard. Such an increased hazard might be due to: a) prolonged retention of the tritium-containing compound, b) enhanced biological effectiveness of the radioactive disintegration due to conversion of the hydrogen atom in a vital molecule to a helium atom (transmutation effect), or c) an enhanced radiation effect due to origin of the beta ray within a vital molecule. If the hydrogen of all molecules in the body were uniformly labeled with tritium, this would add perhaps 50% to the whole body radiation dose from body water alone. Any larger increased radiation dose from organically bound tritium could occur only if tritium were preferentially incorporated or retained, in comparison with ordinary hydrogen. This possibility was reviewed by Weston (1973) who concluded that, "it is apparent that large kinetic isotope effects are often found for tritium-labeled compounds. In tracer experiments utilizing tritium, observed rate constants could easily differ by an order of magnitude from those for the analogous unlabeled compound. If tritium from a source of HTO at constant specific activity is incorporated into a biological system by irreversible chemical reactions, it will be discriminated against; and the tritium level in the biological system will remain lower than that of the source. Conversely, kinetic isotope effects in the back exchange to remove tritium after incorporation will favor retention of tritium in the biological system."

Although rather large isotope effects occur in individual chemical reactions, the overall effects in biological organisms seem relatively small, as discussed by Shtukkenberg (1968). Thompson and Ballou (1954) compared tritium and deuterium in rats, as did Glasscock and Duncombe (1954). The effects were small, as they were in a study of algae (Crespi et al. 1972). It therefore seems reasonable to assume, as was done in the dosimetric calculations for this Statement, that tritium will behave like ordinary hydrogen; any error introduced by such an assumption will probably overestimate the effects of tritium.

The significance of transmutation effects has been a controversial subject, but there now appears to be agreement on the following conclusions, as expressed by Feinendegen and Bond (1973): "The effects of intracellular tritium are overwhelmingly due to beta irradiation of the nucleus. Transmutation effects do not produce a measurably increased effect under most conditions and are detectable only, if at all, under highly specialized laboratory conditions. The origin of tritium beta tracks in, or their close juxtaposition to, the

DNA molecule does not appear to enhance the degree of somatic effects." Studies of the induction of gene mutations in mice also indicate no substantial transmutation effect (Cumming et al. 1974).

Concern has been expressed for the case in which a developing female fetus is exposed to elevated body water levels during oocyte formation; tritium incorporated in these germ cells would be retained until ovulation, and this might constitute a special genetic hazard (Radford 1969). Osborne (1972), however, has estimated that in such a circumstance, less than 0.2% of the initial dose rate to the nucleus originates from tritium incorporated in DNA, and that it would be 30 years before the initial dose from body water was equaled by the cumulative dose from DNA-incorporated tritium.

It would thus appear quite certain that tritium incorporated into organic compounds poses no substantially increased hazard beyond that accounted for by its contribution to whole body dose.

Tritium is a pure beta emitter of very weak energy--18.6 keV maximum. The linear energy transfer (LET) of such a weak beta is higher than that of more energetic beta, x-, or gamma radiation, and much experimental effort has been devoted to determining whether this higher LET is reflected in an increased relative biological effectiveness (RBE). The International Commission on Radiological Protection in its report on Permissible Dose for Internal Radiation (ICRP 1959) used a quality factor of 1.7 for tritium, the value employed in the dosimetric calculations for this Statement. RBE studies were reviewed by Vennart (1968), who concluded "that a value of QF different from unity of either tritium or other β -emitters is hardly justified, and the ICRP reduced the tritium quality factor to unity in 1969, an action concurred in by the National Council on Radiation Protection and Measurements" (1971). More recently, further evidence has been presented to justify a value higher than unity (Johnson 1973 and Moskalev et al. 1973). Of particular interest are studies of Dobson et al. (1974, 1975) on the survival of female germ cells in young mice exposed to a continuously maintained level of tritium oxide in body water. These studies seem to indicate an increasing RBE with protraction of exposure, with the suggestion of a limiting RBE value of about 4 at very low doses. It is important to note, however, that an increasing RBE at very low doses for the relatively high-LET beta radiation from tritium, is (on theoretical grounds, at least) more likely due to a decreased biological effectiveness of the reference, low-LET radiation, than to an absolute increase in tritium effectiveness.

With specific regard to the RBE for genetic effects, the induction of mutations by tritium in mice has been recently studied at Oak Ridge National Laboratory (Cumming et al. 1974). The report of these studies presents the following conclusion: "Thus, if absorbed dose to the testis is accepted as meaningful for purposes of comparison with gamma or X-rays, the . . . point estimate of relative biological effectiveness (RBE) for postspermatogonial germ-cell stages is close to 1, with fairly wide confidence intervals. The point estimate of RBE for spermatogonia is slightly above 2, with confidence intervals which include 1, and there remains the suggestion that the distribution of mutants among the seven loci may differ from that produced by gamma rays" (Cumming et al. 1974).

In summary, it may be concluded that research on both somatic and genetic effects attributed to tritium has failed to produce results markedly different from those which would have been predicted from a general knowledge of ionizing radiation. It may then be assumed that the conventional methods of estimating radiation dose and biological effect, as employed in this Statement, are applicable to tritium.

E.7 SPECIFIC CONSIDERATION OF HEALTH EFFECTS FROM CARBON-14

The radiological significance of ^{14}C has received much attention because 1) carbon occurs everywhere in nature, including man; 2) ^{14}C has a long half-life, 5730 years; and 3) weapons tests have significantly increased global ^{14}C levels (UNSCEAR 1977, pp. 41-42). Only recently has attention been directed to the considerably smaller ^{14}C releases that may be expected from the nuclear fuel cycle (ERDA 1975, Hayes 1977).

As with tritium, there is concern that transmutation effects (i.e., effects resulting from the conversion of a carbon atom to a nitrogen atom in a vital molecule) may increase the health risk from ^{14}C beyond that attributable to the beta-radiation dose. This is of particular concern with regard to genetic effects. Direct experimental data to settle this question are not available. In his original article (1958) calling attention to health risks from ^{14}C , Pauling concluded "that the special mechanism involving ^{14}C atoms in the genes themselves is less important than irradiation in causing genetic damage." Totter, Zelle and Hollister (1958), reviewing the then available data, concluded that "subject to large uncertainty, the transmutation effect of ^{14}C atoms contained in the genetic material of the human body could lead to about the same number of genetic mutations as the radiation effect from ^{14}C ."

The general problem of transmutation effects has received much recent study, and the occurrence and importance of such effects has been clearly demonstrated for ^{32}P (Krisch and Zelle 1969). Less work has been done with ^{14}C , and reported results are not entirely consistent. In studies with Drosophila (fruit flies), Lee and Segal observed little, if any, mutagenic effect from ^{14}C -thymidine incorporated in sperm. They concluded that "if transmutation of ^{14}C is mutagenic at all, it is less effective than ^{32}P (in similar experiments) by two orders of magnitude;" and that, "for practical purposes in considering mutagenic hazards or toxicity effects due to chromosome breakage, only the beta radiation of ^{14}C needs to be considered."

On the other hand, McQuade and Friedkin (1960) observed twice the frequency of chromosome breakage in onion root tips after administering thymidine with ^{14}C -labeling in the methyl group, as with ^{14}C -labeling in the 2 position. This seems to imply a differential transmutation effect, since the labeling position should not influence beta-radiation-induced effects. There is, in any case, no experimental evidence for a transmutation effect that is many times larger than the radiation effect, although such claims have been made on theoretical grounds (Golenetskii et al. 1976). Therefore, based on what appears a preponderance of informed opinion (Krisch and Zelle 1969, and Lee and Segal 1973), this report does not consider the possibility of ^{14}C transmutation effects.

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APPENDIX F

REFERENCE ENVIRONMENT FOR ASSESSING ENVIRONMENTAL IMPACTS

The following reference environment was developed as an aid in assessing environmental impacts associated with construction and operation of waste treatment, interim storage and/or final disposition facilities. The reference environment concept is used to replace, where appropriate, the criteria-type approach to generic environmental assessment.

The reference environment was developed primarily from data on existing plant sites in the midwestern United States. There is, however, no intent to endorse this area or type of environment for any nuclear fuel cycle facility. Since the reference environment is to be used in a generic or hypothetical sense, references supporting the descriptive material were not considered necessary and are not included. The reference environment is representative of the surface geology only and has nothing to do with the deep geology as may be applicable to siting to waste repositories in geologic media.

For assessment of environmental effects, it is assumed that each waste management facility is located (independently, not collocated) within the reference environment. Although an artificiality, analysis of impacts from waste management facilities centered at the same location simplifies calculations and permits direct comparison of impacts among facilities on the same environmental features.

F.1 LOCATION OF SITE

Regardless of the size of the site or purpose to which it is to be put, the center of the site is assumed to be located 8 km west of the R River, about 13 km northwest of Town A in county A, and 50 km northwest of a major metropolitan area (City G) in a midwestern state.

F.2 REGIONAL DEMOGRAPHY AND LAND USE

The reference environment is located in a region that is mainly rural; the land is used chiefly for farming. The nearest communities are A, about 13 km southeast of the site, with a population^(a) of about 2,000; B (population 400) about 6 km northwest; C (population about 1,000) about 8 km east; D (population 1,100) about 16 km southwest; and E (population 3,000) about 16 km south. The closest large cities are F (population 40,000) about 32 km northwest and G (population 1,800,000) about 50 km southeast.

The population within a 1-km radius (300 km²) of the site is about 12,000. Within an 80-km radius of the site (20,000 km²) the population is about 2,000,000, of which about 93% resides in the G metropolitan area (see Table F.2.1).

In County A, and in County B just across the R River to the northeast, about 82% of the land is used for farming. The main crops in these two counties, which include all land

TABLE F.2.1. Projected Year 2000 Population in Reference Environment

RANGE, km	1.6	3.2	4.8	6.4	8.0	16	32	48	64	80	TOTALS
N	0	4	4	18	160	210	1,115	3,641	2,137	1,209	8,498
NNE	0	4	4	14	26	157	986	3,350	4,185	1,872	10,598
NE	0	6	18	72	109	232	1,306	4,897	2,848	6,371	15,859
ENE	0	4	12	72	145	333	2,025	2,677	8,743	6,209	20,220
E	0	4	12	145	537	993	1,321	9,094	6,344	14,195	32,645
ESE	0	4	20	353	118	610	3,400	50,482	123,104	163,155	341,246
SE	0	25	245	1,069	194	632	5,063	46,789	581,389	579,114	1,214,520
SSE	0	4	18	45	157	374	3,466	18,642	59,435	32,445	114,586
S	0	4	41	67	112	1,097	5,438	5,844	10,131	7,334	30,068
SSW	0	15	26	67	126	571	3,177	4,809	6,411	7,317	22,523
SW	0	30	65	58	50	423	1,835	4,656	6,106	6,856	20,079
WSW	0	6	55	93	65	414	3,007	1,901	7,515	4,442	17,498
W	0	9	31	78	73	379	1,730	3,600	3,326	4,805	14,031
WNW	4	8	8	44	29	332	1,662	6,495	6,493	5,984	21,059
NW	0	6	9	21	44	293	5,277	47,196	4,061	4,501	61,408
NNW	0	8	15	55	181	165	1,204	2,753	2,480	4,533	11,394
TOTALS	8	141	583	2,271	2,126	7,215	42,012	216,826	834,708	850,342	1,956,232
CUM TOTAL (rounded)	8	150	730	3,000	5,100	12,000	54,000	270,000	1,100,000	2,000,000	2,000,000

(a) Populations are assumed to be those for the year 2000.

within 16 km of the site, are soybeans, corn, oats, and hay. It is expected that these two counties will remain largely agricultural and that the population distribution will not change significantly with time.

A wildlife refuge is located about 14 km northeast to 19 km north of the site. A state park is located about 10 km west-southwest of the site, and a state forest and campground are about 14 km northeast of the site.

F.3 GEOLOGY

The area in which the reference sites are situated is assumed to occupy a terrace at an elevation of 300 m above sea level (MSL). Several flat alluvial terraces comprise the main topographic features in the vicinity. Many of these terraces are lower than that at the site and lie at an average elevation of 280 m above sea level and, in general, slope away from the river at grades of 2 or 3%. The topography in the area of the site is essentially typical of that in the region.

The rocks that underlie this region are classified as pre-Cambrian and are very old. Glaciation probably less than 1,000,000 years in age, as well as recent alluvial deposition, has mantled the older basement rocks with a variety of unconsolidated materials in the form of glacial moraines, glacial outwash plains, glacial till and river bed sediments. This cover of young soils rests upon a surface of glacially carved deeper rock consisting sequentially in depth of sandstone, shale and granitic rocks. The upper surface of underlying rock can support unit foundation loads up to 73,000 kg/m². The bedrock surface is irregular and slopes generally to the east or southeast.

The nearest known or inferred fault is 37 km southeast of the site. There is no indication that faulting has affected the area of the site in the last few million years. Within the last 100 years, only two earthquakes were recorded as having occurred within 160 km of the site. The first occurred in 1917 and had an intensity of VI on the modified Mercalli scale. The epicenter was located about 100 km northwest of the site. The second occurred in 1950; it had an estimated intensity of V to VI and the epicenter was located about 130 km north-northwest of the site. For construction of facilities in this area the design basis earthquake relates to a horizontal acceleration of 0.1 g.

F.4 HYDROLOGY

Large supplies of ground water are available from the R River outwash plain alluvium, glacial moraine, and from underlying sandstones in the area. The general course of deep ground-water flow is to the southeast. The regional gradient broadly parallels the trend of the topography and the surface drainage. The natural surface drainage of the immediate site area is mainly to the southeast, toward the river.

The R River tributaries close to the site area are S Creek, 8 km northwest, and T Creek, 5 km southwest. The B River flows parallel to and east of the R River, joining the R 24 km downstream from the site area.

The ground-water levels near the site are relatively flat and slope toward the river during normal river stages. During periods of high river flow, there may be some reversal of ground-water flow near the river. These reversals would be of short duration and infiltration of water from the river would be limited. The gradient toward the river is re-established after the high water recedes.

River flow information based on data from the R River gaging station is as follows:

Number of years of record	40
Average annual flow, ℓ /sec	120,000
Minimum recorded flow, ℓ /sec	6,200
Maximum recorded flow, ℓ /sec	1,300,000

River flow and temperature data pertinent to the reference site are shown in Figures F.4.1 and F.4.2, respectively.

Flow duration data for the R River calculated in the vicinity of the reference site are shown in Figure F.4.3. Based on these data, the flow is expected to exceed 50,000 ℓ /sec 90% of the time and 27,000 ℓ /sec 99% of the time.

The average river velocity at the site varies between 0.5 and 0.8 m/sec for flows below 280,000 ℓ /sec. The river drops about 3 m from 2.4 km upstream to 2.4 km downstream of the site. Rapids frequently occur in this stretch of the river.

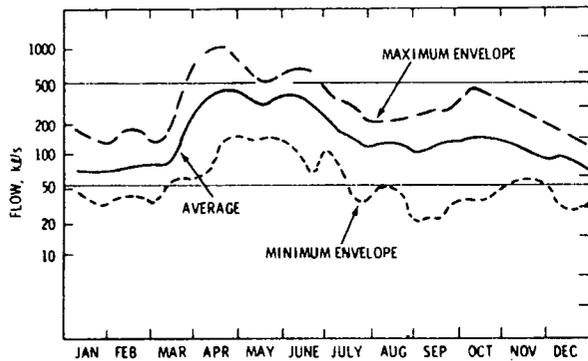


FIGURE F.4.1. Daily Average and Extreme River Flows at the Reference Site

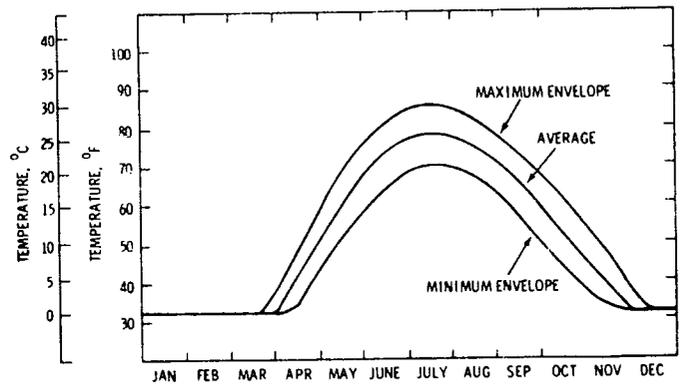


FIGURE F.4.2. Daily Average and Extreme Water Temperatures at the Reference Site

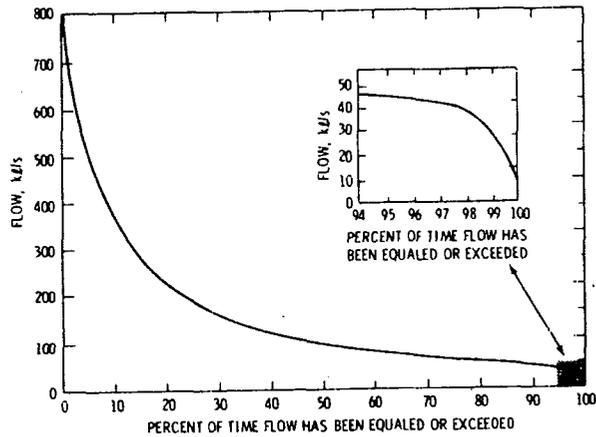


FIGURE F.4.3. River Flow Duration Data for R River at the Reference Site

The 1-in-1000-year flood is expected to reach 281 m MSL (mean sea level), and the maximum flow of record (1965) is estimated to have reached 279 m MSL. Normal river stage in the vicinity of the site is about 276 m MSL, and the site grade is 300 m MSL.

A study was conducted to determine the predicted flood discharge flow and water level at the site resulting from the "maximum probable flood" as defined by the U.S. Army Corps of Engineers. The "maximum probable flood" was estimated as 10 million ℓ /sec with a corresponding peak stage of elevation 286 m MSL at the reference site. The peak level at the site would be reached in about 12 days from the onset of the worst combination of conditions resulting in the "maximum probable flood."

The R River water's chemical characteristics are given in Table F.4.1.

The nearest domestic water supply reservoir is the G Water Works Reservoir. This reservoir is located in northern G and is fed by the R River from an intake about 64 km downstream from the reference site area. (This water supply serves about 1.8 million people)

The ground-water table under normal conditions is higher than the river; thus ground water and runoff drain to the river. There are numerous shallow wells supplying residences and farms along the river terrace. The closest public water supply well is the A city well, which obtains water 72 m below ground level.

TABLE F.4.1 R River Water Chemistry Summary of 12 Monthly Samples

	<u>Minimum</u>	<u>Maximum</u>	<u>Average</u>	<u>Std. Dev.</u>	<u>No.</u>
Solids-mg/l					
Total	143	216	185	23.2	12
Dissolved	125	208	178	27.8	12
Suspended	1.2	18.4	7.5	6.2	12
Hardness-mg/l (As CaCO ₃)					
Total	98	174	147	24.8	12
Calcium	70	120	99	15.6	12
Magnesium	28	58	48	9.9	12
Alkalinity-mg/l (As CaCO ₃)					
Total	91	165	140	24.3	12
Phenolphthalein	0	12	1.8	4.1	12
Gases-mg/l					
Free carbon dioxide	--	--	--	--	--
Ammonia-nitrogen (N)	0.0	0.09	0.02	0.03	12
Anions-mg/l					
Carbonate (CO ₃)	0.0	14.4	2.10	4.96	12
Bicarbonate (HCO ₃)	111	201	166	29.1	12
Hydroxide (OH)	--	--	--	--	--
Chloride (Cl)	0.30	5.00	1.43	1.48	12
Nitrate-nitrogen (N)	0.07	0.55	0.26	0.15	12
Sulfate (SO ₄)	6.3	13.5	9.5	2.2	12
Phosphorus-soluble (P)	0.012	0.057	0.030	0.012	12
Silica (SiO ₂)	3.2	12.5	7.7	3.3	12
Cations-mg/l					
Calcium (Ca)	28.0	48.1	39.7	6.28	12
Magnesium (Mg)	6.8	14.1	11.6	2.4	12
Sodium (Na)	2.8	6.4	5.0	1.1	12
Total iron (Fe)	0.04	0.52	0.23	0.13	12
Total manganese (Mn)	--	--	--	--	--
Potassium (K)	--	--	--	--	--
Miscellaneous					
Color (APHA units)	20	80	39	22	12
Turbidity (JTU)	1.00	4.50	2.53	1.48	12
Ryznar index (AT 77°F)	6.64	7.86	7.21	0.377	12
Conductivity (mmho)	192	350	292	49.8	12
pH	7.40	8.60	8.15	0.308	12
BOD (mg/l)	0.9	2.5	1.4	0.58	12
Dissolved oxygen (mg/l)	8.0	15.0	10.6	2.1	11
Temp. (DEG. C)	0.0	23.0	9.69	9.03	12

F.5 METEOROLOGY

The general climatic regime of the site is that of a marked continental type characterized by wide variations in temperature, scanty winter precipitation, normally ample summer rainfall, and a general tendency to extremes in all climatic features. Temperature data, obtained by adjusting 54-year climatological summaries for G and B, indicate that January is the coldest month, with average daily maximum, mean and minimum temperatures of -6, -11 and -16°C, respectively. July is the warmest month, with corresponding temperatures of 28, 22, and 16°C. Table F.5.1 shows monthly statistics.

The number of days with maximum temperatures of 32°C and above is estimated to be 12. The numbers of days with a minimum temperature of 0°C or below and -18°C or below are estimated to be 168 and 40, respectively. The January relative humidities at 7:00 a.m., 1:00 p.m., and 7:00 p.m., EST, are estimated to be 76, 68, 70%, respectively. The corresponding humidities for July are 86, 55, and 55%. Monthly average humidities are shown in Table F.5.2.

The annual average rainfall is about 76 cm. The maximum 24-hr total rainfall for the period 1894-1965 for B was 13 cm and occurred in May. Thunderstorms have an annual frequency of 36 and are the chief source of rain from May through September. Snowfall in the area has an annual average of 110 cm, with occurrences recorded in all months except June, July and August. The extremes in annual snowfall of record are a 15-cm minimum and a 220-cm maximum.

Annually, the winds are predominantly from the northwest or from the south through southeast. This bimodal distribution is characteristic of the seasonal wind distributions as well. The average windspeed for spring is 11 km/hr and for the other seasons about 16 km/hr. The maximum reported windspeed of 160 km/hr, reported in July 1951, was associated with a tornado. Tornadoes and other severe storms occur occasionally. Eight tornadoes were reported in the period 1916 to 1967 in county A. The theoretical expected frequency of a tornado striking a given point in this area is 5×10^{-4} per year. For design purposes a maximum windspeed of 580 km/hr is assumed to be associated with tornadoes.

TABLE F.5.1. Monthly Temperature Statistics (°C)

	<u>Jan</u>	<u>Feb</u>	<u>March</u>	<u>Apr</u>	<u>May</u>	<u>June</u>	<u>July</u>	<u>Aug</u>	<u>Sept</u>	<u>Oct</u>	<u>Nov</u>	<u>Dec</u>
Maximum	-6.1	-4.4	3.3	12.8	20.0	25.0	28.3	26.7	22.2	15.0	4.4	-3.3
Minimum	-16.1	-14.4	-6.7	1.7	7.8	13.3	16.1	15.0	10.0	3.9	-4.4	-12.2
Mean	-11.1	-9.4	-1.7	7.2	13.9	18.9	22.2	21.1	16.1	9.4	0.0	-7.8
Extreme Max	15.0	16.1	27.8	32.8	40.6	39.4	41.7	40.0	40.6	32.2	23.9	17.2
Extreme Min	-38.9	-36.7	-34.4	-15.6	-5.7	6.0	5.6	3.3	-5.6	-13.3	-27.8	-33.9

TABLE F.5.2. Mean Monthly Relative Humidity percent

<u>Jan</u>	<u>Feb</u>	<u>Mar</u>	<u>Apr</u>	<u>May</u>	<u>June</u>	<u>July</u>	<u>Aug</u>	<u>Sept</u>	<u>Oct</u>	<u>Nov</u>	<u>Dec</u>
74	75	73	66	62	66	68	70	70	66	73	78

It is estimated that natural fog restricting visibility to 0.4 km or less occurs about 30 hr/year. Icing due to freezing rain can occur between October and April, with an average of one to two storms per year. The mean duration of icing on utility lines is 36 hr.

Diffusion climatology comparisons with other locations indicate that the site is typical of the region, with relatively favorable atmospheric dilution conditions prevailing.^(a) Frequency of thermal inversion is expected to be about 32% of the year, and the frequency of thermal stabilities is 19% slightly stable, 27% stable, 20% neutral, and 34% unstable. The joint distribution of windspeed, direction, and stability is given in Table F.5.3.

TABLE F.5.3 Annual Average Joint Frequency Distribution, Percent of Occurrence

WIND SPEED(M/S)	STABILITY TYPE	WIND DIRECTION															
		NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	
1,10	A	.02	0.00	.01	0.00	.01	.01	0.00	.02	.02	0.00	0.00	.01	.02	0.00	0.00	
2,50	A	.10	.11	.17	.12	.07	.15	.11	.15	.25	.21	.31	.35	.32	.37	.19	
4,30	A	.27	.31	.21	.22	.22	.30	.47	.58	.40	.58	.42	.41	.62	.77	.62	
6,50	A	.06	.07	.01	.06	.25	.72	.73	1.38	.62	.16	.15	.09	.51	.64	.59	
9,10	A	0.00	0.00	0.00	0.00	0.00	.02	.32	.31	.07	0.00	.04	.04	.10	.17	.14	
12,20	A	0.00	0.00	0.00	0.00	0.00	0.00	.02	.04	0.00	0.00	0.00	0.00	0.00	0.00	0.00	
1,10	B	.02	.02	.02	.04	.02	.02	.02	.06	.01	.04	.04	.01	0.00	.01	0.00	
2,50	B	.09	.04	.16	.05	.12	.09	.06	.15	.11	.14	.12	.16	.14	.16	.10	
4,30	B	.19	.14	.05	.14	.05	.09	.14	.23	.21	.19	.07	.23	.16	.22	.22	
6,50	B	.01	.01	.02	.01	.07	.07	.05	.07	.02	0.00	0.00	.06	.21	.21	.17	
9,10	B	0.00	0.00	0.00	0.00	.01	0.00	.01	.02	.01	0.00	0.00	.01	0.00	.11	.01	
12,20	B	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	
1,10	C	.02	.01	.04	0.00	0.00	.01	.01	.01	.01	0.00	.05	.01	.02	0.00	0.00	
2,50	C	.05	.10	.04	.06	.07	.07	.06	.06	.12	.06	.09	.10	.07	.09	.09	
4,30	C	.11	.04	.07	.04	.07	.05	.17	.12	.05	.04	.02	.16	.17	.22	.14	
6,50	C	.02	.01	.04	.01	.04	.05	.04	.07	.01	.01	.02	0.00	.17	.15	.07	
9,10	C	.05	0.00	0.00	0.00	0.00	0.00	0.00	.02	0.00	0.00	.01	.02	.02	.15	0.00	
12,20	C	.01	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	
1,10	D	.19	.16	.22	.04	.07	.09	.07	.10	.10	.14	.10	.11	.07	.10	.09	
2,50	D	.54	.41	.45	.42	.47	.46	.47	.53	.32	.32	.41	.67	.47	.99	.90	
4,30	D	.73	.60	.62	1.03	.94	1.17	.90	.66	.49	.36	.36	.72	1.30	1.65	1.30	
6,50	D	.21	.27	.19	.46	.61	.61	.37	.35	.20	.22	.10	.38	1.24	1.40	.78	
9,10	D	.10	.04	0.00	0.00	.01	.04	.07	.05	.02	.01	.01	.16	.07	.21	.10	
12,20	D	.02	0.00	0.00	0.00	0.00	0.00	0.00	.02	0.00	0.00	0.00	0.00	0.00	.01	0.00	
1,10	E	.06	.02	.11	.15	.07	.20	.09	.14	.11	.10	.10	.12	.05	.17	.00	
2,50	E	.51	.35	.25	.61	.57	.72	.33	.30	.21	.42	.52	.65	.49	.75	.48	
4,30	E	.27	.10	.06	.34	.26	.91	.69	1.16	.57	.49	.30	.67	.64	.78	.35	
6,50	E	.01	0.00	0.00	.11	.01	.15	.30	.56	.30	.10	.06	.05	.22	.09	.02	
9,10	E	0.00	0.00	0.00	0.00	0.00	0.00	0.00	.16	0.00	0.00	0.00	0.00	0.00	0.00	0.00	
12,20	E	0.00	0.00	0.00	0.00	0.00	0.00	0.00	.01	0.00	0.00	0.00	0.00	0.00	0.00	0.00	
1,10	F	.07	.11	.11	.10	.12	.06	.12	.19	.10	.15	.15	.21	.09	.10	.20	
2,50	F	.14	.22	.15	.42	.40	.46	.54	.65	.32	.35	.35	.63	.37	.67	.36	
4,30	F	0.00	.01	0.00	.02	.05	.20	.30	.48	.27	.14	.05	.10	.04	.17	.14	
6,50	F	0.00	0.00	0.00	0.00	0.00	0.00	.04	.01	.05	0.00	0.00	.01	0.00	0.00	0.00	
9,10	F	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	
12,20	F	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	
1,10	G	.09	.04	.12	.20	.11	.19	.32	.31	.17	.31	.46	.33	.22	.20	.12	
2,50	G	.05	.05	.07	.14	.17	.32	.65	.74	.21	.23	.28	.28	.15	.35	.36	
4,30	G	.01	.01	0.00	.01	.01	.01	.02	.16	.02	0.00	.02	.01	0.00	.12	.04	
6,50	G	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	
9,10	G	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	
12,20	G	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	
SUMMARY DATA FOR STABILITY TYPE		A		B		C		D		E		F		G			
TOTAL PERCENT FOR STABILITY IS		17.85		5.90		4.00		34.37		19.46		10.44		7.94			

(a) An investigation of the variations in atmospheric dispersion among a number of sites around the nation was made to determine differences to be expected in radiation dose calculations based on atmospheric dispersion because of different synoptic conditions for different locations. For five of the eight sites studied it was determined that the maximum atmospheric dispersion coefficient at 1100 m and at 72 km from the point of release was not greater by more than a factor of two over that of the reference site. It was no greater than a factor of six for any of the other three sites studied.

F.6 PATHWAY PARAMETERS RELEVANT TO RADIOLOGICAL DOSE CALCULATIONS

Radiation exposure of man via airborne pathways may include that from radiation emitted from overhead plumes and ground-level clouds; direct radiation from radionuclides deposited on the ground; inhalation of radionuclides released to the atmosphere; and consumption of foods produced from vegetation upon which radionuclides have been deposited or which have been grown in soils on which deposition has accumulated. Such foods may include vegetables from local gardens; milk from cows foraging on pasture grass; or meat from animals raised on pasture and feed grown in the vicinity of the plant. These pathways are illustrated in Figure F.6.1.

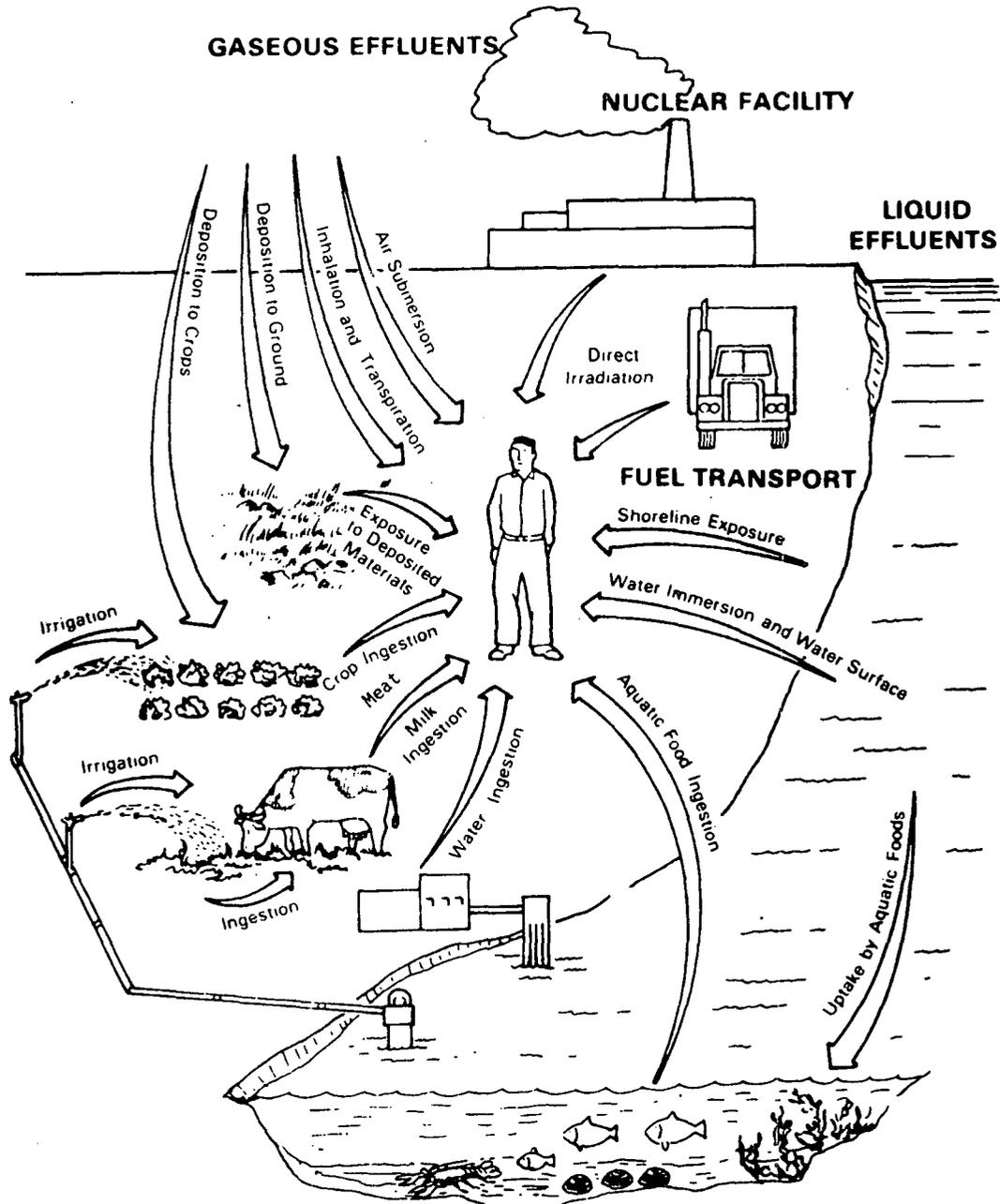


FIGURE F.6.1. Pathways for Radiation Exposure of Man

Also, as illustrated in Figure F.6.1, radiation exposure of man via surface water pathways may include that from ingesting radionuclides with drinking water, consumption of aquatic foods, and direct radiation from surface waters received through shoreline activities or swimming or boating.

For the milk and home garden pathways, the nearest dwelling is assumed to be a farmhouse adjacent to the site boundary southeast of the main plant where the maximum ground-level atmospheric dispersion factor (\bar{x}/Q') is about 3×10^{-7} sec/m³ for ground-level releases and 1.5×10^{-8} sec/m³ for tall stack releases. A milk cow is assumed to be kept at this farm and maintained on fresh pasture 7 months of the year. It is assumed that a garden is kept for vegetables; however, there are no large truck gardens in the area.

For the farm-crop-irrigation pathway, it is assumed that about 82% of land in the vicinity of the site is farmed. Production is essentially 60% soybeans (0.7 kg wet weight/m²) 30% corn, oats and other grain (0.35 kg wet weight/m²) and 10% hay (1.5 kg wet weight/m²). For dose calculation purposes, it is assumed that 10% of the average flow rate ($\sim 12,000$ l/sec) of the R River in the vicinity of the plant site is drawn from the river during June, July and August for irrigation of 250 km².

For the recreational and aquatic food pathways, it is assumed that in the vicinity of the plant a "maximum-exposed individual"^(a) may spend 100 hr/yr swimming or boating and may spend 500 hr/yr obtaining 10 kg of fish and 10 kg of fresh water mollusca. Aquatic foods are assumed to be consumed within 24 hours of the time they are harvested.

For pathways to the population, it is assumed that 85% of the 2 million residents within 80 km of the site obtain their drinking water from the R River. Travel time to the consumer from a point on the river nearest the site is taken to be 48 hours. It is assumed that on the average each person will spend 5 hr/yr swimming and 10 hr/yr boating or fishing downstream from the site. The average per capita fish consumption for this area has been estimated to be 1.1 kg/yr. It is assumed that 10% of this consumption is from fish obtained downstream from the site.

(a) A "maximum-exposed individual" is an individual whose habits tend to maximize his or her dose.

APPENDIX G

REFERENCE SITES FOR ASSESSING SOCIAL AND ECONOMIC IMPACTS

A generic assessment of socioeconomic impacts incorporates the assumption that various sites may be under consideration for development of nuclear waste management facilities. Since the potential sites may differ considerably in their distinguishing characteristics (e.g., population size, composition, and distribution; industrial composition of the labor force; and availability of social services) it is necessary to examine the potential effects of energy facilities on several alternative sites. For example, it is reasonable to assume that a highly urbanized community offering a wide range of services to residents will experience fewer negative effects from the construction and operation of a project than will a sparsely populated rural community. In the latter, even a relatively small project could produce disruptive effects.

In addition to considering alternative reference sites, it is also necessary to assess the effects of several types of nuclear waste management facilities. These facilities differ substantially in terms of the length of time and the number of workers needed for construction, the number of workers required for planned operation, the potential hazards created through storage and transportation of noxious materials, and the amount of land occupied. Thus, it is reasonable to expect that the variety and degree of socioeconomic impacts will differ according to the facility in question.

Each of the three reference sites utilized in the assessment of social and economic impacts is based on realistic conditions chosen on the basis of criteria listed below. They should not be construed to represent an endorsement of any specific site for facility location. Since the reference sites are to be used in a generic or hypothetical sense, source references supporting the descriptive material are presented in terms of their broad, general areas rather than in specific terms (see Table G.2.1). One of the three reference sites coincides with the reference environment described to in Appendix F.

G.1 CRITERIA FOR REFERENCE SITE SELECTION

To permit an assessment of a wide range of variation in impacts, three reference sites were selected for analysis from a larger number of possible locations for nuclear waste facilities on the basis of two criteria:

- population size. The three sites vary markedly in terms of the total number of inhabitants at the site and in the surrounding region.
- population distribution. The three sites exhibit variations in population density and degree of urbanization.

G.2 CHARACTERISTICS OF REFERENCE SITES

To emphasize that the reference sites are hypothetical, they are simply labeled Midwest, Southeast, and Southwest. Each reference site consists of a single county. The region within which the county is located is defined as the aggregation of all counties falling substantially within a 50-mile radius of the facility. If more than half of a county is included within that 50-mile radius, it is included in the region.

Regional populations are important for assessing site impacts because a sizable portion of the site labor force may commute to work from regional localities. Fifty miles represents the maximum commuting distance that most workers are willing to undertake. Furthermore, population redistribution within the region may result in project-related impacts.

Table G.2.1^(a) summarizes data for the site counties and surrounding regions. Two types of comparisons can aid in the interpretation of these data. First, there are marked differences among the sites, whether based on county or regional comparisons. Second, there are important differences between the county and the region for each site. From the population data it is evident that the Southeast and Midwest regions are highly urbanized when compared with the Southwest region. Differences among the three counties are even greater. While the Midwest site falls within the most urbanized region, the county containing that site has the smallest urban component. In fact, each site county is less urbanized than its corresponding region, reflecting the likelihood that waste repositories will be situated away from urban centers and densely settled areas. The density figures also support this observation.

The sites vary dramatically in terms of population change over the 1965 to 1970 period, with the Southwest site showing a marked decline, the Midwest site a comparable increase, and the Southeast site remaining relatively stable. From 1970 to 1975 all sites gained population, and the differences among the rates of change are smaller than in the preceding 5-year period. These changes over the decade can be attributed to two components: natural change and net migration. Natural change is the difference between births and deaths. Net migration is the difference between the number of persons moving into an area and the number moving out. Each site has experienced an excess of births over deaths, thus serving to moderate the population loss due to emigration from the Southwest and Southeast sites over this period while increasing the growth experienced by the Midwest site. Population change has important consequences in the capacity of a site to absorb impacts. Counties that are experiencing rapid population growth may be more likely to plan to accommodate further demand on local services than counties that are not growing. On the other hand, counties that are losing population may have under-utilized service sectors, which would then be available to serve the needs of project-related immigrants.

While the Southeast county has a high urban component compared with the Midwest county, the Southeast county is only one-fifth as densely populated as the Midwest county. In the

(a) The population data used here are based on realistic locations covering the period 1970 to 1975. Analyses of future impacts are based on projections of these data to the year 1980 and beyond.

Table G.2.1 Selected Data characteristics of Three Reference Sites,
Socioeconomic Impact Analysis (a)

Characteristic	Southwest Site		Midwest Site		Southeast Site	
	County	Region	County	Region	County	Region
<u>Population</u>						
Estimated total population 1975	42,000	142,000	47,000	2,154,000	17,000	487,000
% Change 1965-1970	-8.5	-8.6	15.0	11.1	-1.4	4.2
% Change 1970-1975	3.2	5.8	24.9	3.8	11.9	2.6
Unemployed construction labor force, 1980	--	390	--	10,660	---	2,420
Net migration rate 1965-1970	-14.9	-14.6	7.4	3.0	-6.6	-2.4
Net migration rate 1970-1975	-0.9	0.5	18.4	-0.7	6.1	-2.2
% Urban 1970	76.9	78.9	8.4	85.1	40.9	50.1
Density 1970 (persons per sq. mi.)	9.9	9.2	57.8	246.8	31.1	60.1
% Nonwhite 1970	2.9	5.0	0.3	2.4	41.3	38.3
% Families with children under 18, 1970	56.8	59.3	59.4	59.7	57.6	57.6
Median age 1970	27.2	26.3	25.6	25.6	24.9	24.5
<u>Employment</u>						
Nonworker to worker ratio	1.7	1.7	1.6	1.3	1.4	1.5
% Employed in farming	5.7	5.8	13.6	2.2	8.5	4.9
% Employed in construction	7.7	5.6	6.0	3.7	5.2	5.8
% Unemployed	5.1	5.1	4.5	3.3	4.6	4.3
% Below poverty level	17.8	16.6	10.8	5.5	24.6	22.3
Median family income	7,870	7,965	8,936	11,242	6,997	7,166
<u>Education</u>						
Median years school completed	11.9	12.0	12.2	12.3	9.8	10.6
% High school graduates	49.3	51.3	56.0	64.5	29.8	37.0
<u>Housing</u>						
% Housing units renter occupied	25.9	25.7	15.8	31.5	33.4	33.2
% Units vacant	16.1	18.2	6.4	3.4	9.4	8.3
Trailers as % of housing units	2.5	3.3	6.5	1.8	7.2	5.8
% Units lacking plumbing	5.0	3.6	8.7	4.0	29.3	19.7
% Units built 1939 or earlier	19.2	17.6	53.3	41.1	36.8	30.6
% Units with 1+ persons per room	11.7	11.6	9.5	6.9	15.1	13.1
% Units using public sewer service	77.8	82.1	39.3	82.7	45.8	46.3

(a) These data were developed from standard sources, but since sites are generic, no identifying information is given.

Southwest region most people live in towns just large enough to qualify as urban by the U.S. Census Bureau (2500 or more). The nearest metropolitan center (population 50,000/year or more) is over 100 miles from any part of the Southwest region. The Midwest region, however, contains a very large metropolitan center, though the site itself is primarily rural.

Looking briefly at the data related to employment, it is apparent that the Midwest site residents enjoy the highest standard of living. This is true for both the county and the region and is reflected by relatively high family income, low percent unemployed, and low percent below the poverty level, defined for 1975 by the U.S. Census Bureau as \$5500 for a nonfarm family of four. In contrast, almost one-quarter of the Southeast site residents are below the poverty level, and the median income for the Southeast region is less than two-thirds that for the Midwest region. Similar regional differences are reflected in the data presented on education. The Southeast site residents are substantially less educated than residents from the other two sites, a condition to be expected from the more rural character of the Southeast site.

Housing variables are critical because they reflect the ability of a community to adequately accommodate a substantial population influx. Vacancy rates coupled with the condition of housing determine the ease with which the incoming labor force can find adequate, affordable living space. In this regard, the Southwest site is apparently best situated to accommodate a population influx. It has a higher vacancy rate and substantially newer housing units in better condition when compared with the other two sites. In addition, a very high proportion of the Southwest site's housing facilities are connected to a public sewer service.

The three reference sites selected are each distinct in terms of demographic, economic, and social service characteristics. The relative size and significance of socioeconomic impacts that might accrue from the construction and operation of waste management facilities will be conditioned in large part by these characteristics of the reference site.

APPENDIX H

HAZARD INDICES

The total quantity of radioactive material to be isolated can be compared to the isotope quantities that naturally occur in the earth's crust (Winegardner and Jansen 1974, Smith 1975). This comparison can be used to indicate the relative hazard that may result from the burial of radioactive waste (i.e., geologic isolation). Early efforts to develop safety perspectives on geologic isolation led to the development of hazard indices. These indices attempted to combine those parameters that characterize waste isolation into an index on public health and safety. The indices use one or more of the following parameters: quantity of radioactive material, specific activity, decay properties, chemical and physical form, packaging, toxicity, time behavior, and pathways.

Some hazard indices that have been developed are listed and defined in Table H.0.1. Studies in which they have been used include: the comparison of the toxic content of high-level waste to the toxic content of the uranium ore and tailings from which it came (Cohen 1976, 1977); the comparison of the toxic level of Pu sent to high-level waste against the toxic level of lead sent to waste (Cohen 1975); The Reactor Safety Study (NRC 1975) (risk of nuclear plant accidents compared to risk of natural disasters); risk of plutonium shipments (Hall et al. 1977); risk of natural and man-caused radioactivity (Turnage 1976); the relevance of nuclide migration at Oklo (Walton and Cowan 1975); underground testing of nuclear devices (Teller et al. 1968); direct impact of disruptive events (Starr 1970); and risk comparisons to alternative energy resources (Grahn 1976, pp. 371-387; Straker and Grady 1977; Cottrell 1976; Blot et al. 1977; Starr et al. 1972; Petrikova 1970; McBride et al. 1977).

The various hazard indices attempt to incorporate additional considerations (such as the concentration of the waste material and the pathways for the nuclear material to enter the biosphere) into the comparison between nuclear waste and naturally occurring radioactive materials. As can be seen in Table H.0.1, the total quantity of radioactive material (Q), the maximum permissible concentration (MPC), and the maximum permissible intake (MPI) give measures of the toxicity of the waste material. A better index of the toxicity of the material is the hazard measure (HM) (Walton and Cowan 1975), which is the quantity of water required to dilute the material to its acceptable maximum permissible (non-toxic) concentration. Thus, the HM is a number that is proportional to the toxicity of the waste material. The "first modified hazard measure" (HM1) (Walsh et al. 1977) compares the anticipated exposure (or dose) to an allowable limit. It was introduced to evaluate the effect of environmental pathways on hazards from a variety of environmental pollutants including nuclear wastes. The second modified hazard measure (HM2) (McGrath 1974) is a measure of the potential hazard of radioisotope releases in air and water. It is a number proportional to such hazard. The third modified hazard measure (HM3) (Petrikova 1970) is a quantity to assess the radioactive risk to future generations from future releases of radioisotopes. It is the

TABLE H.1. Hazard Indices^(a)

Hazard Index	Definition and Inputs	Interpretation (for Nuclear Waste Isolation) ^(b)
Quantity of Radioactive Material (Q)	Waste Inventory (or waste released)	Comparison of waste inventories to natural radionuclides (or for use below) (Winegardner and Jansen 1974).
Maximum Permissible Concentration (MPC)	10 CFR 20	Relative hazards of radioactive species (or for use below).
Maximum Permissible Intake (MPI)	$MPI_{air} = (7300 \text{ m}^3/\text{yr}) (MPC_{air})$ $MPI_{water} = (0.8 \text{ m}^3/\text{yr}) (MPC_{water})$	Same as MPC.
Hazard Measure (HM)	$HM = Q/MPC$	Volume of air or water to dilute Q radionuclides to one MPC. (Winegardner and Jansen 1974, Smith 1975)
Modified Hazard Measure (HM1) (Walsh et al. 1977)	$HM1 = D/D_2$ D = exposure D ₂ = exposure limit	Ratio of anticipated exposure to allowable limit.
Modified Hazard Measure (HM2) (McGrath 1974)	$HM2 = Q(a/MPI_{H_2O} + b/MPI_{air})$ a, b = fractions of Q released to water and air	$HM2 = Q(a/MPI_{H_2O} + b/MPI_{air})$
Modified Hazard Measure (HM3) (Smith and Kastenbergh 1976)	$HM3 = \int_t^{t+d} (Q(t')/MPI) dt'$	Number of MPI in the environment versus time.
Potential Hazard Measure (PHM) (Gera and Jacobs 1972)	$PHM = P \frac{Q}{MPI} \frac{1}{\lambda}$ P = probability of reaching man λ = decay constant	Risk of releasing Q versus time.
Hazard Index (HI) (Claiborne 1975, Haug 1977)	$HI = \frac{Q}{MPC(V)}$ V = entrained volume	Number of MPCs per unit volume.
Hazards Available Index (HA) (Bruns 1976)	$HA = \log_{10} HI + \log_{10} TF$ TF = transport factors	HI with pathway transport efficiency included.
Isolation Time (T) (Voss and Post 1976)	$T = -\frac{1}{\lambda} \ln \frac{MPC V_f D}{A L}$ V _f = groundwater volume flow rate D = dilution factor A = waste leach area L = leach rate	Time which nuclides must be held to reduce concentration to one MPC.
Relative Toxicity Index (RTI) (Haug 1977, Hamstra 1975, Haug 1976, Cohen and Tonnessen 1977, Roehing 1977)	$RTI = \frac{(Q/MPC)_{waste}}{(Q/MPC)_{U_{ore}}}$	Ratio of HI of the waste to HI of the uranium ore mined to generate the waste. This has been generalized to compare with substances other than uranium.

(a) A compilation from published studies.

(b) As defined by originator.

number of MPI in the environment versus time. The potential hazard measure (PHM) (Gera and Jacobs 1972) is an index that is proportional to the quantity of radionuclides buried as a function of time and modified by the probability that this material will reach man. The hazard index (HI) (Claiborne 1975) is a quantity that is proportional to the specific toxicity of a radionuclide. It was formulated to assess the benefits of actinide removal from high-level waste. The hazards available index (HA) (Bruns 1976) is a modification of the hazards index that includes a pathways transport efficiency. It has been used to compare the hazard from Purex waste to the hazard from fallout. The isolation time (T) (Voss and Post 1976) is the time radionuclides must be held to limit their concentration in ground water to one MPC. It was introduced to characterize the effectiveness of geologic isolation in restraining the transport of radionuclides via the groundwater transport path. The relative toxicity index, RTI (Haug 1977, Hamstra 1975, Haug 1976, Cohen and Tonnessen 1977, Rochlin 1977), is the ratio of the hazard indices of nuclear waste to uranium ore. This index has been generalized to compare to toxicity of nuclear waste to the toxicity of other naturally occurring toxic elements.

Although each hazard index has merit for a particular set of conditions, the provision of simple measures of hazard can confuse rather than clarify. For this reason hazard indices are infrequently used in this Statement and dose and associated health effects are presented instead.

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APPENDIX I

COMPARISON OF DEFENSE PROGRAM WASTE TO COMMERCIAL RADIOACTIVE WASTE

Repositories for commercial high-level and TRU wastes may also be used for disposal of defense program wastes.^(a) This appendix provides a comparison of defense program radioactive wastes with commercial radioactive wastes. These comparisons indicate that both the HLW and the TRU defense-program wastes could be accommodated in repositories designed for disposal of commercial wastes with comparable environmental impact.

I.1 HIGH-LEVEL WASTE COMPARISONS

The waste quantities and radionuclide contents of defense program and commercial high-level wastes (HLW) are compared in Table I.1.1. The estimated quantities of defense program high-level waste are based on the assumption that waste forms having a 25% loading of waste oxides are encapsulated in 0.6-m x 3-m (2-ft x 10-ft) canisters that are filled to 80% of capacity. The commercial HLW is assumed to be contained in canisters that are 3 m (10 ft) long with diameters up to 0.3 m (1 ft). The quantity of commercial HLW in individual canisters is adjusted, either by dilution or by varying canister diameter, to meet the allowable heat output imposed by the disposal system. The radionuclide content and heat output of individual defense program HLW canisters is a factor of 5 to 10 or more below that of the commercial HLW canisters. The radionuclide content in the defense program HLW canisters relative to the commercial HLW canisters ranges from about the same magnitude for plutonium to orders of magnitude less for some of the other nuclides.

(a) President Carter, Feb. 12, 1980.

TABLE I.1.1. Comparison of Defense and Commercial High-Level Waste

	Canisters Required	Heat Output, kW/Canister ^(a)	Radionuclide Content, Ci/Canister ^(a)					
			⁹⁰ SR	¹³⁷ CS	²³⁸ Pu	²³⁹ PU	²⁴¹ Am	²⁴⁴ Cm
Defense HLW ^(b)								
• Savannah River	8.0×10^3	0.2	1.5×10^4	1.5×10^4	1.4×10^2	2.9	8.2	8.2
• Idaho Falls	1.2×10^4	0.09	7.3×10^3	7.4×10^3	4×10^1	4×10^{-1}	6.0×10^{-1}	3.1×10^{-1}
• Hanford	2.6×10^4	0.06	5.2×10^3	4.8×10^3	2×10^{-2}	9.2×10^{-1}	6.5	5.4×10^{-1}
Total	4.6×10^4							
Commercial HLW ^(c)								
	1.0×10^5	3.2	1.4×10^5	2.0×10^5	1.8×10^2	4.3	1.7×10^3	1.4×10^4
	to	to	to	to	to	to	to	to
	2.8×10^5	1.2	5.0×10^4	7.1×10^4	6.5×10^1	1.5	6.1×10^2	5.1×10^3

(a) Nominal values, assuming uniform distribution of waste radionuclides among the canisters.

(b) Estimated data for the year 1990. Treated waste volumes (assuming a waste form having a 25% loading of waste oxides) and radionuclide contents supplied by J. L. Crandall and W. R. Cornman of the High-Level Waste Lead Office at Savannah River. Canister requirements based on 0.6-m-diameter x 3-m-long canisters, 80% full of treated waste. Heat outputs based on the contained radionuclides.

(c) Data from this Statement for the reprocessing of spent fuel containing 2.4×10^5 MTHM (Case 3) and radioactivity at 6.5 years after reactor discharge. Canister requirement dictated by the heat output allowed by the disposal system.

I.2 TRU WASTE COMPARISONS

The defense program TRU wastes will require a variety of treatment procedures. Because potential treatment procedures for these wastes are not yet sufficiently well defined to develop good estimates of treated waste forms and quantities, they are compared to commercial TRU wastes on the basis of untreated quantities and radionuclide compositions in Table I.2.1. The quantity of defense program TRU wastes is about the same magnitude as the estimated commercial wastes for the Case 3 growth assumptions (see Chapter 7). The plutonium content is similar to the commercial waste. In both cases, the americium and curium content varies over a wide range.

TABLE I.2.1. Comparison of Defense and Commercial TRU Wastes

	Volume, m ³				TRU Content, Ci/m ³ (a)					
	Retrievably Stored	Buried	Soil Contaminated by Burial	Total	kg	kg/m ³	²³⁸ Pu	²³⁹ Pu	²⁴¹ Am	²⁴⁴ Cm
Defense TRU Waste										
On hand as of September 30, 1979										
• Hanford	8.0 x 10 ⁴	1.6 x 10 ⁵	1.4 x 10 ⁵	3.8 x 10 ⁵						
• INEL	3.7 x 10 ⁴	5.6 x 10 ⁴	0	9.3 x 10 ⁴		8 x 10 ⁻³	5 x 10 ⁻¹	3.5 x 10 ⁻¹	1.4	3.4 x 10 ⁻²
• LASL	1.5 x 10 ⁴	1.1 x 10 ⁴	1.7 x 10 ⁴	4.3 x 10 ⁴						
• ORNL	1.2 x 10 ³	6.1 x 10 ⁴	1.6 x 10 ⁵	1.7 x 10 ⁵						
• SRP	2.4 x 10 ³	2.7 x 10 ⁴	3.4 x 10 ⁴	6.3 x 10 ⁴		2.8 x 10 ⁻²	1.5 x 10 ²	1.2		1.6 x 10 ¹
• Other	2.4 x 10 ³	5.7 x 10 ³	5.0 x 10 ³	1.3 x 10 ⁴						
Total	6.5 x 10 ⁴	2.6 x 10 ⁵	3.6 x 10 ⁵	7.6 x 10 ⁵	>1.1 x 10 ³	>1.4 x 10 ⁻³				
Estimated Annual Generation, 1980 to 2000										
	6.8 x 10 ³	0	0	6.8 x 10 ³						
Commercial TRU Waste										
Estimated to Result from Reprocessing 2.4 x 10 ⁶ MTHM (Case 3 growth projection)										
• Untreated from FRPs	7.0 x 10 ⁵	0	0	7.0 x 10 ⁵	1.1 x 10 ⁴	1.5 x 10 ⁻²	7.9	5.5 x 10 ⁻¹	4.8 x 10 ⁻¹	9.0 x 10 ⁻¹
• Untreated from MOX-FRPs	6.6 x 10 ⁴	0	0	6.6 x 10 ⁴	4.5 x 10 ³	6.8 x 10 ⁻²	1.6 x 10 ¹	1.2	1.2 x 10 ²	0
Total	7.7 x 10 ⁵			7.7 x 10 ⁵	1.6 x 10 ⁴					

1.4

(a) Composition of defense TRU waste is based on estimate for retrievably stored waste only as of late 1977.

Sources of Data for Table I.1.1:

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APPENDIX K

GEOLOGIC REPOSITORY DESIGN CONSIDERATIONSK.1 THERMAL CRITERIA

A major factor in geologic isolation of radioactive waste is the heat generated by high-level waste (HLW) or spent fuel assemblies. This heat flows from the waste, through the emplaced canister and other protective material, into the host rock formation, through the rock surrounding or overlying this formation, and eventually out into the atmosphere. The heat can have definite impacts on:

- the integrity and recoverability of the waste canisters
- room and pillar stability
- integrity of the waste form over long periods of time
- the integrity of the host rock and the surrounding rock units
- any overlying aquifers and buoyancy effects on ground-water flow
- long-term uplift and subsidence of overlying rock.

To assure that the impact of the heat on these factors will not be detrimental to waste isolation objectives, a systematic determination of the repository design thermal loads is required that includes:

- establishment of limits for conditions affected by heat
- determination of acceptable thermal loads that will not bring about conditions beyond the assigned limits
- development of repository design thermal loads, taking into account safety, engineering and operational requirements.

Design limits for the repository can be specified in terms of temperature and thermo-mechanical criteria. Preliminary estimates of acceptable thermal conditions are summarized in Table K.1.1 and discussed below.

- Maximum Uplift Over Repository

Uplift over the repository centerline was chosen as a measure of the far-field structural consequences of repository thermal loading. The 1.2 to 1.5 m of maximum uplift, neglecting subsidence, is based on the assumption that rock-mass movements caused by uplift may be no worse than movements caused by subsidence over mines in sedimentary rocks, which are sometimes more than twice the stated limit. Far-field effects are currently being studied to determine whether 1.2 to 1.5 m of uplift is reasonable. This tentative limit may change as more information is developed. In any case, this limit must be reevaluated for each site so that the effects of rock-mass movement on the hydrological regime and long-term safety may be assessed.

TABLE K.1.1. Thermal and Thermomechanical Limits for Conceptual Design Studies

Event	Limits
Far-Field Considerations	
Maximum uplift over repository	1.2 to 1.5 m (Russell 1977)
Temperature rise at surface	0.5°C (Science Applications, Inc. 1976)
Temperature rise in aquifers	6°C (Science Applications, Inc. 1976)
Near-Field Considerations	
Room closure during ready retrievability period--salt	10 to 15% of original room opening (Russell 1977)
Room stability--granite, basalt rock strength-to-stress ratio	2 within 1.5 m of openings (Dames and Moore 1978)
Room stability--shale with continuous support rock strength-to-stress ratio	1 within 1.5 m of openings (Dames and Moore 1978)
Pillar stability--non-salt strength-to-stress ratio	2 across mid-height of pillar (Dames and Moore 1978)
Very-Near-Field Considerations	
Maximum HLW temperature as vitrified waste	500°C (Jenks 1977)
Maximum spent fuel pin temperature	300°C (Blackburn 1978)
Maximum canister temperature	375°C (Jenks 1977)
Maximum rock temperature	250°C to 350°C
Maximum fracture of non-salt rock	15 cm annulus around canister (Russell 1977)

- Temperature Rise at the Surface

Temperature rise at the surface has been limited to < 0.5°C to avoid undesirable effects on the biota. This limit must also be reevaluated for each site (Science Applications, Inc. 1976).

- Temperature Rise in Aquifers

Temperature rise in aquifers has been limited to < 6°C because the flow velocity could conceivably carry the higher-temperature water outside the repository area. In addition, temperature rise and temperature gradients can influence ground-water flow patterns and, in the worst case, may provide a transport mechanism to return nuclides to the biosphere. This limit is currently under study and must be reevaluated for each site, with consideration given to flow rate, salinity, and geochemistry, including dissolution, transport, and

subsequent precipitation of minerals. Permissible temperature rises of 8° and 28°C for stagnant aquifers 30 and 90 m deep, respectively, have also been proposed (Science Applications, Inc. 1976).

- Near-Field Considerations

Rooms must be accessible at the end of the retrievability period to allow safe entry for the removal of canisters with the same equipment used to emplace them. Calculated room closures of less than the limit imply that the repository will generally remain structurally stable throughout the retrieval period, although some local failure controlled by local rock conditions not accounted for in the analysis may occur.

In addition to thermal loading, the closure of rooms in a salt repository will depend on the depth of the repository; this relates directly to stress and mine-geometry parameters such as the percent extraction of salt and pillar width-to-height ratios. Room closure calculations appear to be relatively insensitive to stratigraphy provided that the salt near the burial horizon is at least hundreds of feet thick.

- HWL Temperature for Glass, 500°C

Typical borosilicate waste glasses have a transition temperature of about 500°C, with a slightly higher softening temperature. Migration of heavy, separate phases in the glass might occur above the softening temperature. Significant increases in cracking and in leach rates have been observed in test glasses heated for a few months in the range 500° to 800°C. Additional information is available for solid waste temperatures of glass, calcine, and sintered glass ceramic (Jenks 1977, Mendel et al. 1977).

- Spent Fuel Pin Temperature, 300°C

A study of possible failure mechanisms during dry storage of spent fuel assemblies sealed in carbon steel canisters recommended a maximum allowable cladding temperature of 380°C based on stress rupture considerations. Some uncertainty regarding possible stress corrosion cracking was noted. To be safe, a 300°C maximum fuel pin temperature is specified here.

- Canister Temperature, 375°C

Austenitic stainless steel, probably 304L, proposed to be used in HLW canisters undergoes changes in structure during long-term exposure in air at temperatures in the range 400 to 900°C. The observed effect is an increased susceptibility to stress cracking when the steel is subsequently exposed to aqueous solutions (Jenks 1977).

- Rock Temperature, 250°C to 350°C

Behavior of salt deposits at temperatures up to 250°C are believed to be predictable. Laboratory tests (Jacobsson 1977) indicate that unconfined rock-salt samples from several locations begin to decrepitate (disaggregate) in the 260° to 320°C range, but samples from other locations show no decrepitation when heated to 400°C. Decrepitation is undesirable because it reduces thermal conductivity of the salt in the vicinity of a waste package and could lead to undesirable higher temperatures in the container and waste. In the case of bedded salt, decrepitation may release brine, which is also undesirable.

For the other rock media, i.e., granite, shale, and basalt, a good basis for specifying maximum rock temperatures had not been established at the time of this analysis. It (the criteria) will probably be quite site-specific. For shale a 250°C maximum may be reasonable and for the hard rocks temperatures higher than 350°C may be acceptable.

It must be emphasized that the limits shown in Table K.1.1 are based on the best available data at this time. As such, they should be reevaluated as more data become available. In addition, these limits require evaluation on a site-specific basis.

K.1.1 Calculation of Acceptable Thermal Loads

For convenience, the thermal criteria, subsequent analyses, and results are classified into three categories: far-field, near-field, and very-near-field. The far-field refers to the formation at distances far removed from the repository. The near-field represents the region within the repository horizon in the vicinity of the emplacement rooms and associated pillars. The very-near-field refers to the waste package and the rock within a few feet of the canister.

The heat induced into the repository and surrounding formation depends upon repository design and the thermal loadings of the repository. These loadings include: 1) the average waste loading of the repository (averaged over full waste emplacement area) that determines the temperature rise of the formation in the far-field; 2) the local thermal loading (average amount of waste emplaced per unit storage area of the repository) that most directly determines the near-field rock thermal and thermomechanical environments; and 3) individual canister loadings that most directly influence the temperatures in the waste, the canister, and the rock in the immediate vicinity of the waste canister, i.e., in the very-near-field. For a given repository design, acceptable loadings can be determined once appropriate temperature and thermomechanical limits have been established.

Thermal and thermomechanical analyses have been performed to determine acceptable thermal loading values for spent fuel repositories and HLW repositories in salt, granite, shale, and basalt. These studies use an iterative technique that integrates the waste and canister temperature criteria, room and pillar stability analyses, and far-field thermal and rock mass response analyses.

For isolation of HLW, the following steps were followed in the iterative analysis:

- Step 1: Select thermal and thermomechanical criteria.
- Step 2: Propose a conservative room and pillar design without consideration of an imposed thermal loading.
- Step 3: Make near-field heat-transfer calculations to determine the areal thermal loading range of interest.
- Step 4: Make very-near-field heat-transfer calculations to generate very-near-field temperature profiles as a function of areal thermal loading and canister loading.

- Step 5: Make near-field rock mechanics calculations to determine the areal thermal loading that assures room and pillar stability.
- Step 6: Determine maximum canister load from Step 4 data for the areal thermal load from Step 5.
- Step 7: Make far-field thermal and rock mechanics calculations to assure that far-field design limits are not exceeded.

If any of the tentative limits in Table K.1.1 are exceeded in any of the above steps, the previous steps are revised and repeated until the calculational results indicate that the limits are not exceeded.

For spent fuel repository analyses, the above procedure was modified slightly. Because it was decided to place PWR or BWR spent fuel assemblies in individual canisters, the thermal load for a given canister was determined, and Step 6 above was not required. Steps 1 through 3 were followed by Steps 5 and 7. Very-near-field heat transfer calculations were then performed to determine if canister or spent fuel temperature limits were exceeded.

This iterative procedure results in baseline thermal load design values for the canisters in terms of kW per canister at waste emplacement and for the loading of a repository room (local areal thermal load) in kW/acre. The canister load must be sufficiently low so that the waste and canister temperatures do not exceed the values in Table K.1.1. The local areal thermal load must be sufficiently low so that rock mechanics analyses predict room and pillar stability throughout the readily retrievable period, and so that near-field hydraulic conductivities are not significantly increased and long-term as well as far-field restrictions are not exceeded.

The design thermal limits generated by these analyses depend strongly upon characteristics of the repository site and formation. These characteristics include media strength, stress-to-strain ratio, heat capacity, thermal conductivity, overlying strata and their characteristics, etc. The following simplifying assumptions were made for these analyses:

- Only high-level waste and spent-fuel canisters are considered.
- The entire repository is assumed to be loaded simultaneously and instantaneously.
- Thermal properties of geologic media and other materials are based on reasonable estimates.
- The effects of stress upon thermal properties are not included.
- The presence of water is neglected in the thermal analysis.
- Only simplified horizontal stratigraphies are assumed.
- No compaction or subsidence of the formation is considered.

The analyses utilize cylindrical symmetry to describe the temperatures within the waste package. Details of the waste package including overpack and other contents of the emplacement hole are taken into account. In the case of spent-fuel canisters, details of the assemblies, radiation and convection are explicitly included in the calculation. The

boundary conditions at the emplacement hole surface are provided in a three-dimensional Cartesian near-field model with asymmetric spacings between canisters. The heat-generating waste and waste canister are explicitly described as well as the properties of the rock in the pillars and above and below the waste storage room. The storage room was not modeled since it has little impact on canister temperatures. The storage room including radiative and convective heat transfer effects has been included in other calculations, however. The boundary conditions above and below the storage room and canister are provided in a far-field model. Temperatures in this model are calculated in cylindrical symmetry and stratigraphy of the host formation can be explicitly modeled.

The thermal load limits and the controlling factors associated with each limit generated by these analyses for 10-year-old spent fuel and HLW are presented in Table K.1.2. The far-field average repository loading limits are based on the far-field studies and the estimated maximum uplift of the formation caused by heat from the stored waste. Far-field average repository thermal loading limits apply to the thermal density of wastes averaged over each waste type's overall emplacement area, including corridors and ventilation drifts and excluding the areas for shafts or emplacement areas for other waste types. In linear thermomechanical expansion studies for salt, a surface uplift of 1.2 to 1.5 m was obtained for average far-field loadings shown in Table K.1.2. This maximum uplift is felt to be acceptable for a repository at 600 m over the time frame involved (Russell 1977). Similar calculations for granite and basalt for loadings of 190 kW/acre, and shale for 120 kW/acre, give less than 0.4 m of surface uplift. Although Table K.1.2 indicates that thermal loading limits for both the far-field and near-field for spent fuel and HLW in granite, shale, and basalt, and for HLW in salt are equivalent, the far-field average repository loading will always be less because of the passive regions of the repository such as corridors and waste handling areas.

The near-field local areal loading limits are based on room and pillar stability considerations. Near-field local thermal loading limits are applied to the thermal density of wastes in an individual waste type's emplacement room area including the area of one-half the rock pillar on each side. Areas for corridors, shafts, and other waste type emplacement areas, are excluded. Linear thermomechanical analyses based upon the predicted near-field temperature distributions indicate that readily retrievable operations could continue in the storage rooms for at least 5 years with the loadings in Table K.1.2 (Dames and Moore 1978).

Although salt can accept 150 kW/acre based on room and pillar stability considerations, this density cannot be achieved in the case of spent fuel because of the more limiting far-field criteria. Reduced loadings are necessary here because of the long-term heat contributions from the plutonium as shown in Table K.1.3. The additional long-term heat contribution of the plutonium does not affect room stability but does increase surface uplift. In order to meet the far-field limit of 60 kW/acre, the maximum near-field density that can be achieved is 75 kW/acre for spent fuel. All other wastes may be emplaced at the 150 kW/acre near-field and far-field criteria for nonplutonium wastes in salt.

TABLE K.1.2. Thermal Load Limits for Conceptual Repository Designs

	Thermal Load Limit (controlling factor) ^(a)			
	Salt	Granite	Shale	Basalt
Canister Limits During Retrieval Period (kW) ^(b)				
Vitrified glass HLW	3.2(A)	1.7(A)	1.2(A)	1.3(A)
Calcined HLW	2.6(A)	1.6(A)	1.1(A)	1.1(A)
Near Field Local Areal Thermal Loading Limits ^(c) (kW/acre)				
5-yr retrieval--HLW	150(B)	190(B)	120(B)	190(B)
5-yr retrieval--spent fuel	(e)	190(B) ^(f)	190(B) ^(f)	190(B)
Far-Field Average Repository Thermal Loading ^(d) Limits (kW/acre)				
HLW	150(C)	190(B)	120(B)	190(B)
Spent fuel	60(C)	190(B)	120(B)	190(B)

(a) Controlling factors: A = Canister temperature limit
B = Room closure
C = Earth surface uplift.

(b) Analysis assumes 15-cm annulus of crushed rock around waste package.

(c) Acreage includes rooms and adjacent pillars, but not corridors, buttress pillars, and receiving areas.

(d) Acreage includes storage area for waste including corridors and ventilation drifts, but does not include area for shafts, or storage areas for other waste types if separate.

(e) In salt, the emplacement of spent fuel and HLW with plutonium is controlled by the more restrictive 60 kW/acre far-field thermal limit. Otherwise the near-field limit would be 150 kW/acre.

(f) In order to maintain spent fuel cladding temperatures within the 300°C limit with these areal thermal loadings, the annulus around the canister is left open (no backfill). Heat is transferred across this air space more readily than through crushed backfill material and results in cooler canister and cladding temperatures.

TABLE K.1.3. Cumulative Heat Generated by 10-Yr-Old Spent Fuel and High-Level Waste

Years	kW-yr/MTHM	
	Spent Fuel	HLW
	Once-Through Cycle	U & Pu Recycle
0	0	0
10	9	9
50	40	30
100	58	36
200	78	43
300	92	46
400	102	49
500	116	50
1000	143	55

In the very-near-field analyses, the baseline canister-emplacment design was a single overpacked canister placed in a hole. In general, the void space between the sleeve and the hole was assumed to be backfilled with crushed rock. In each of the HLW calculations, a 15-cm annulus of crushed or fractured rock was assumed.

K.1.2 Thermal Loadings Achieved in Conceptual Repository Designs

Engineering or operational constraints may restrict any of the thermal loadings discussed in the above section to values lower than the limits presented in Table K.1.2. These constraints include such factors as reasonable HLW concentration in canisters, available canister sizes, permissible hole spacing, and room stability limitations on hole arrangements. Spent-fuel canister loading is limited in this Statement to a single PWR or BWR spent fuel assembly so that canister heat loads are below limiting values. The HLW canister diameters are reduced as necessary in each case so that the canister loadings are below the limits of Table K.1.2. Alternatively the waste could be diluted with inert material without reducing canister sizes to achieve the same result.

As a hedge against uncertainties in the criteria and other factors and to ensure a conservative estimate of repository capacities, the design areal thermal loadings for both spent fuel and HLW were established at 2/3 of this limiting areal loading parameter in Table K.1.2. The age of both the spent fuel and HLW were assumed to be 6.5 years. Using the criteria in Table K.1.2 for 6.5-year-old waste provided a further degree of conservatism since the criteria were developed for 10-year-old waste (the thermal limits could be increased for younger wastes). The resulting thermal densities actually achieved in the first conceptual repositories are listed in Table K.1.4. The limiting thermal parameter, i.e., near-field or far-field, is denoted by an asterisk. In the case of BWR fuel in shale and the RH-TRU waste in all media except salt, structural limitations on canister placements limit thermal loading.

Temperature profiles calculated for the conceptual repositories using the achieved loadings are shown in Figures K.1.1 through K.1.8. The profiles show temperature increases above ambient temperature as a function of depth at several times after the repository is loaded, for both spent fuel and HLW and for the four geologic media. For example, the profiles for a spent fuel repository at a depth of 600 m in salt with the average loadings of Table K.1.4, are shown in Figure K.1.1. The figure shows that the temperature at the repository depth reaches a maximum value about 70 years after emplacement. The calculation is made assuming that the heat source is uniformly dispersed at the repository level. The temperature is calculated along a line perpendicular to the plane of the repository and passing through the center of the emplacement area. Actual temperatures in the vicinity of the repository level will vary with the discontinuities of the temperature profile around each canister.

Figure K.1.2 gives the profiles for the repository in salt for the high-level waste from the reprocessing cycle. Corresponding profiles for each cycle are shown in Figures K.1.3 and K.1.4 for granite, K.1.5 and K.1.6 for shale, and K.1.7 and K.1.8 for basalt repositories.

TABLE K.1.4 Thermal Loadings Achieved at Conceptual Repositories

<u>Cycle</u>	<u>Thermal Loading at Emplacement</u>	<u>Salt</u>	<u>Granite</u>	<u>Shale</u>	<u>Basalt</u>
<u>Once-Through</u>					
	PWR				
	kW/can	0.72	0.72	0.72	0.72
	Near-field local kW/acre	50	130*	80*	130*
	Far-field average kW/acre	40*	100	65	100
	BWR				
	kW/can	0.22	0.22	0.22	0.22
	Near-field local kW/acre	50	130*	55	130*
	Far-field average kW/acre	40*	100	44	100
<u>U & Pu Recycle</u>					
	HLW				
	kW/can	3.2	1.7	1.2	1.3
	Near-field local kW/acre	100*	130*	80*	130*
	Far-field average kW/acre	76	95	60	95
	RH-TRU (hulls)				
	kW/can	0.32	0.32	0.32	0.32
	Near-field local kW/acre	100*	93	42	77
	Far-field average kW/acre	76	70	32	60

* Denotes limiting thermal parameter.

Predicted temperature histories over the first 100 years for the waste (center line) or spent fuel (center pin), the canister wall, and for the rock near the surface of the emplacement hole are shown for the design canister loadings in Figures K.1.9 through K.1.16. These temperatures correspond to the highest values obtained anywhere in the formation rock. The temperatures have been calculated in models with detailed treatment of the very-near-field, including 15 cm of crushed formation material between the rock and the canister in the emplacement hole. Additional details of the models and analyses are contained in DOE/ET-0028. The results for PWR spent fuel canisters and the HLW canisters, respectively, in a salt formation are shown in Figures K.1.9 and K.1.10. The corresponding temperature histories for granite, shale and basalt are shown in Figures K.1.11 and K.1.12, K.1.13 and K.1.14, and K.1.15 and K.1.16 respectively.

The temperature histories are all well within the temperature criteria in Table K.1.1 except for the center pin temperature for spent fuel in basalt, which just reaches the 300°C criteria. One method of reducing these temperatures is elimination of the crushed backfill surrounding the emplaced canisters. Heat is transferred across the resulting air space more readily than through the crushed backfill material and results in cooler canister and cladding temperatures. A higher conductivity backfill material could also be used.

A tabulation of the material properties used in making these thermal calculations is shown in Tables K.1.5 and K.1.6.

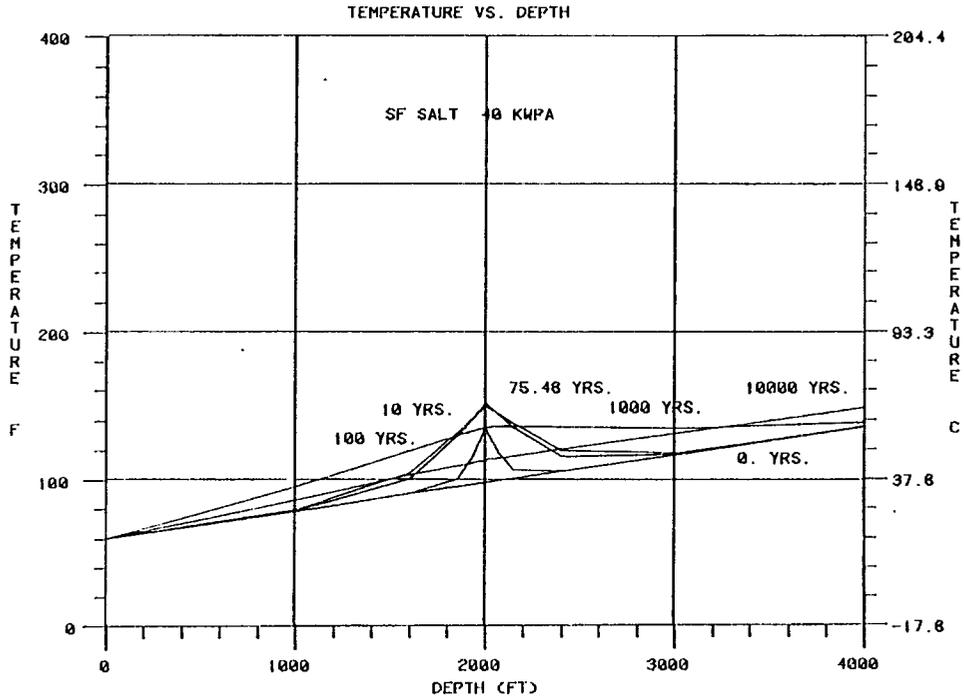


FIGURE K.1.1. Formation Temperature versus Depth and Time for Repository in Salt--Once-Through Fuel Cycle

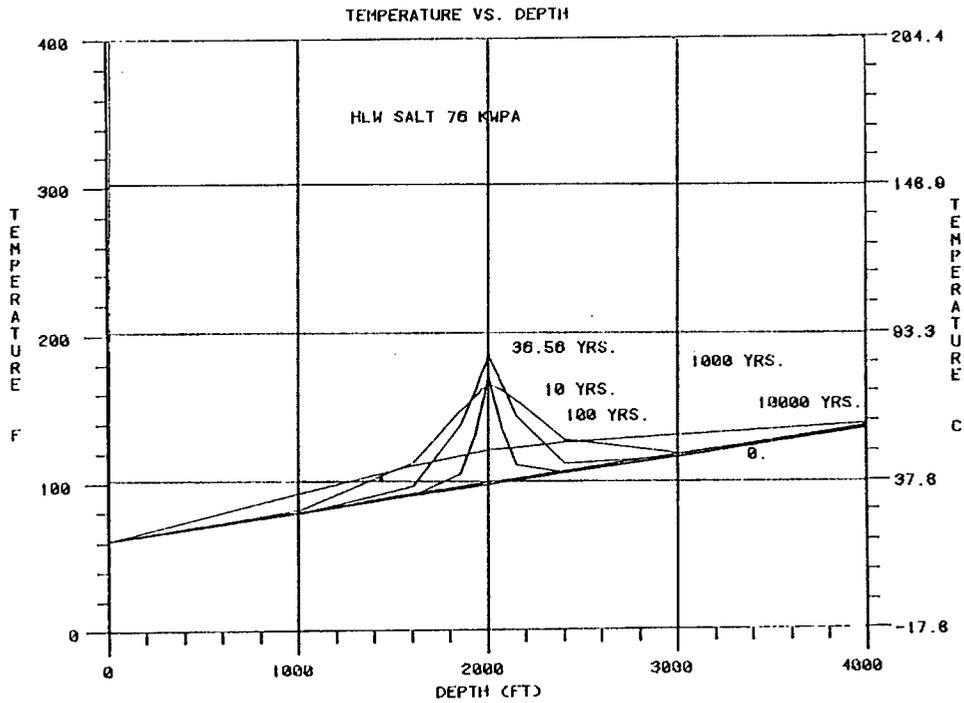


FIGURE K.1.2. Formation Temperature versus Depth and Time for Repository in Salt--Reprocessing Fuel Cycle

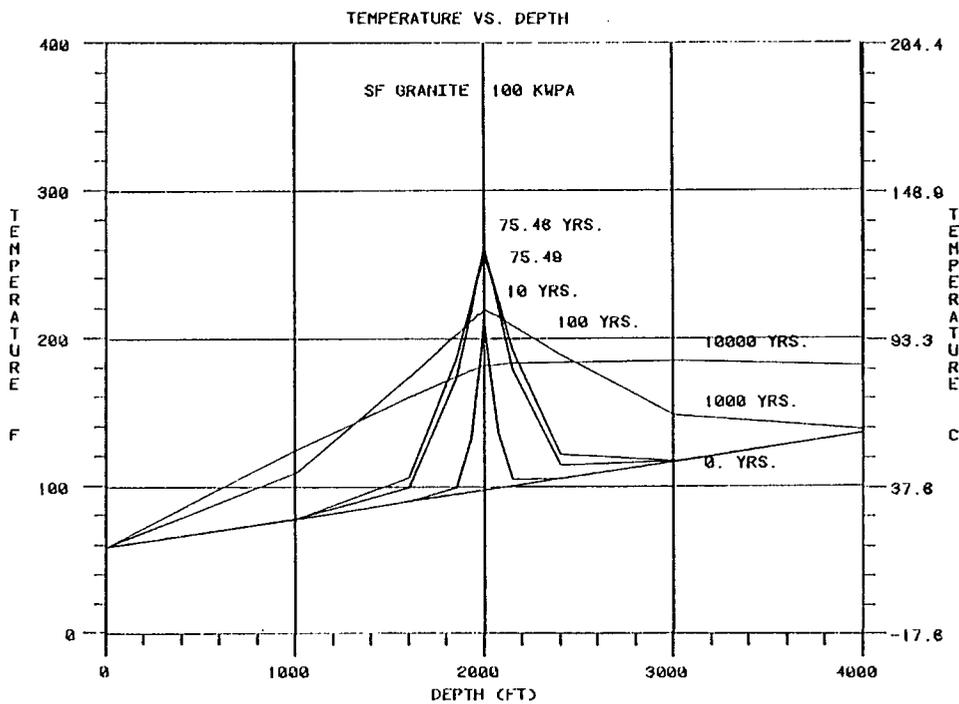


FIGURE K.1.3. Formation Temperature versus Depth and Time for Repository in Granite--Once-Through Fuel Cycle

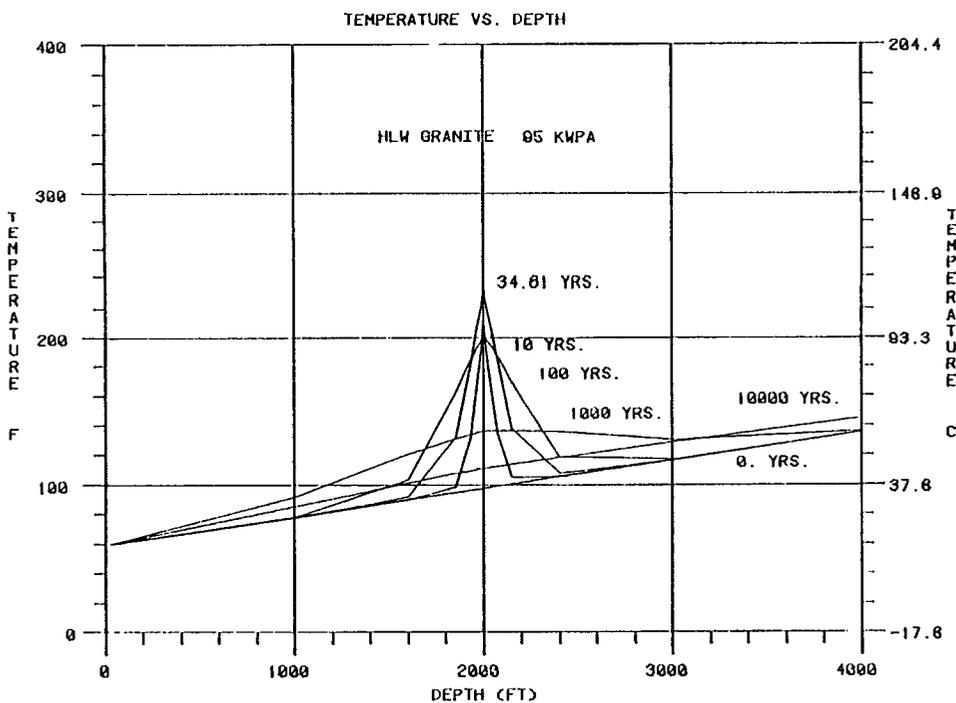


TABLE K.1.4 Formation Temperature versus Depth and Time for Repository in Granite--Reprocessing Fuel Cycle

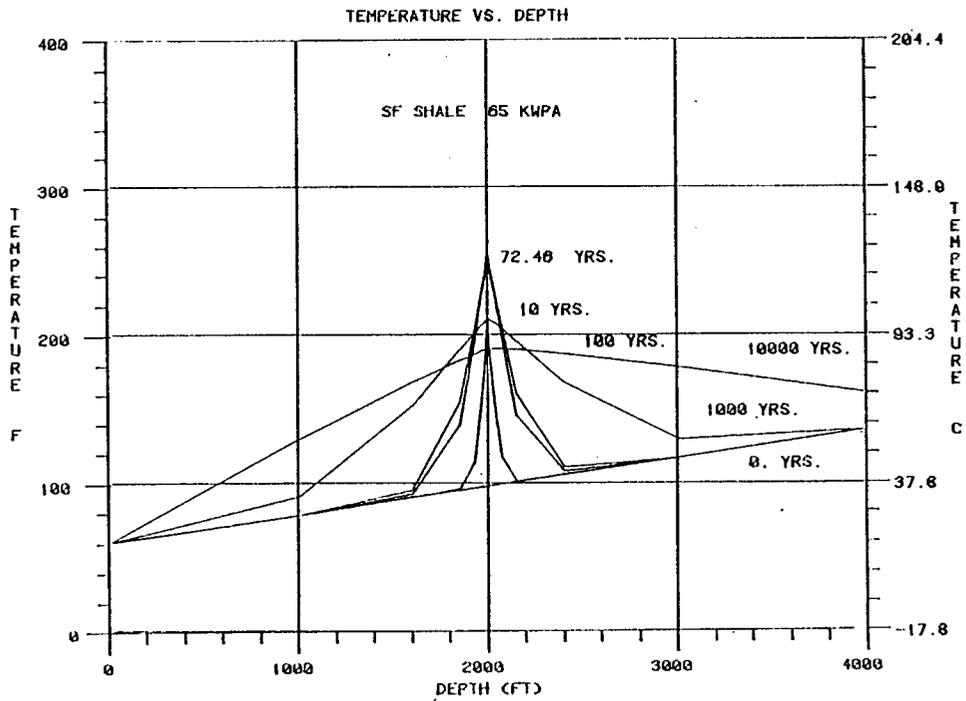


FIGURE K.1.5. Formation Temperature versus Depth and Time for Repository in Shale--Once-Through Fuel Cycle

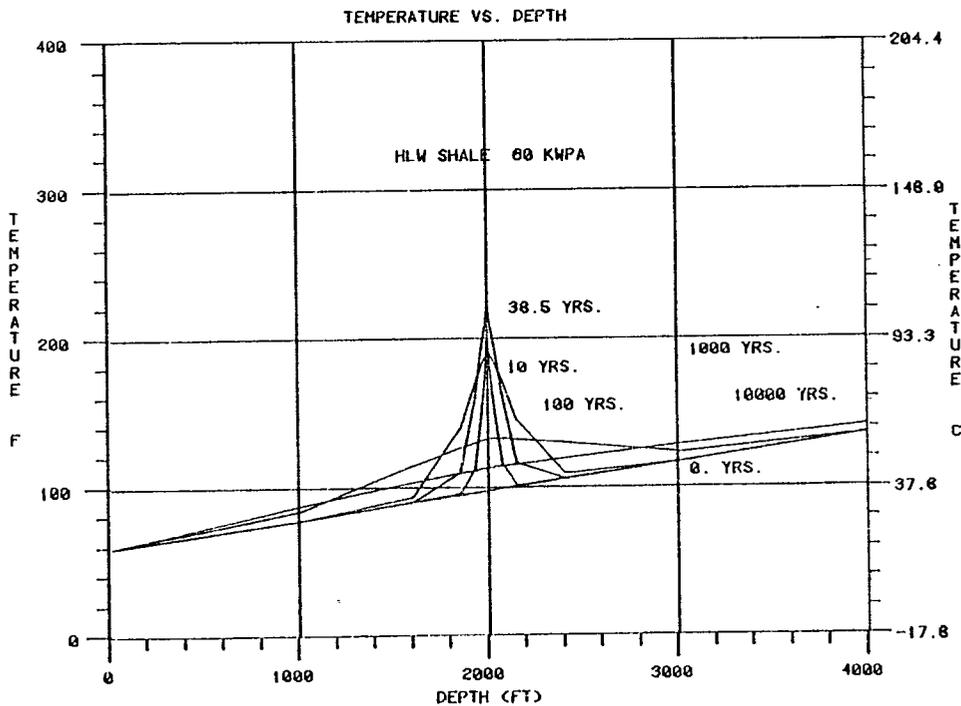


FIGURE K.1.6. Formation Temperature versus Depth and Time for Repository in Shale--Reprocessing Fuel Cycle

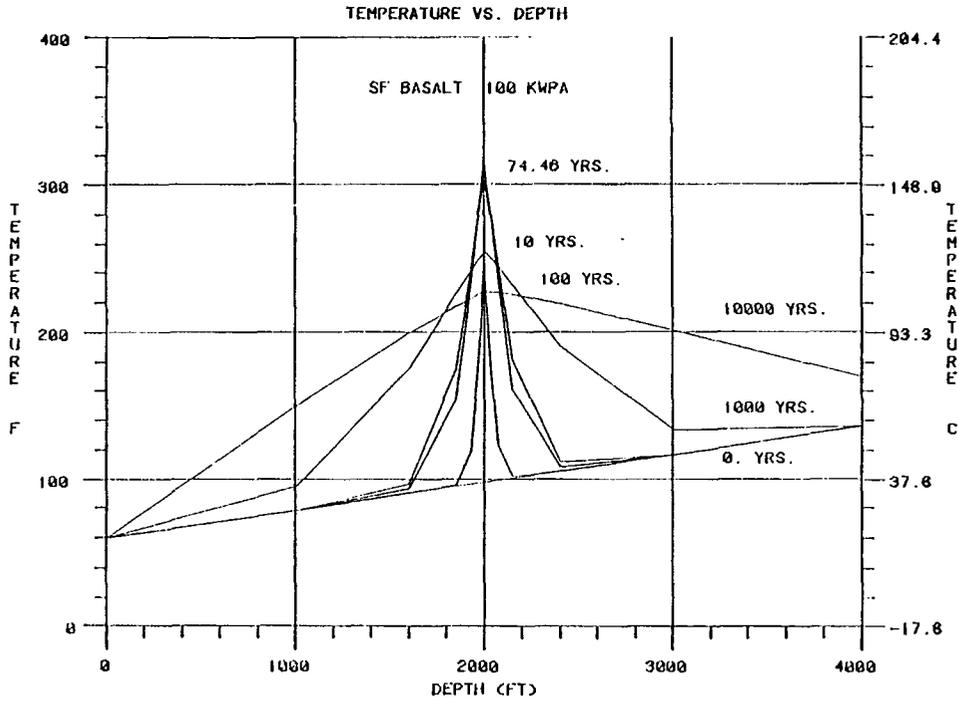


FIGURE K.1.7. Formation Temperature versus Depth and Time for Repository in Basalt--Once-Through Fuel Cycle

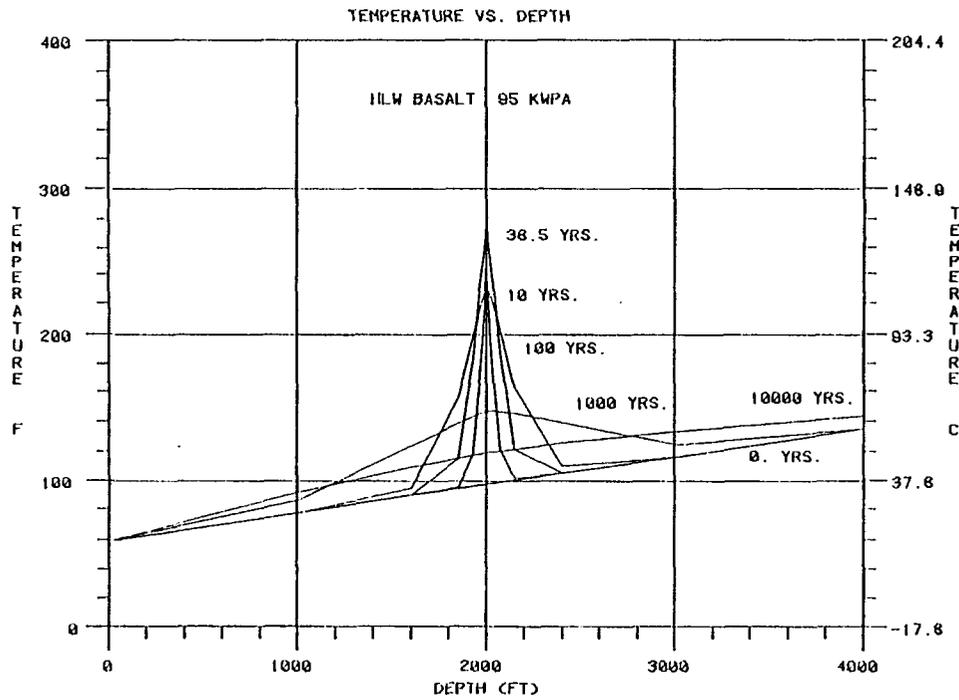


FIGURE K.1.8. Formation Temperature versus Depth and Time for Repository in Basalt--Reprocessing Fuel Cycle

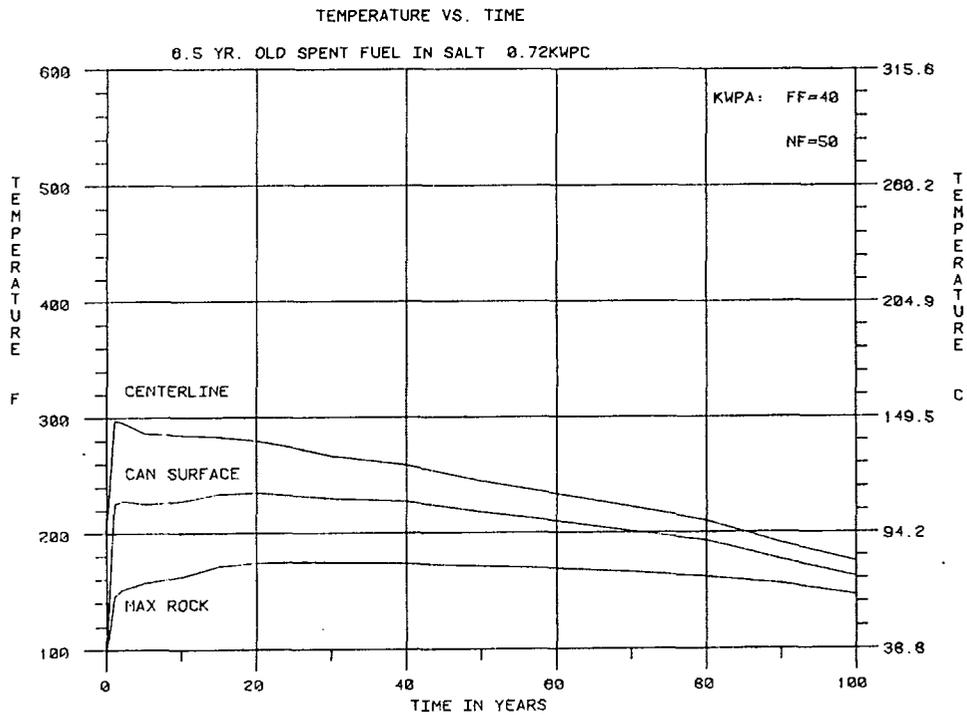


FIGURE K.1.9. Very-Near-Field Temperatures versus Time for Repository in Salt--Once-Through Fuel Cycle

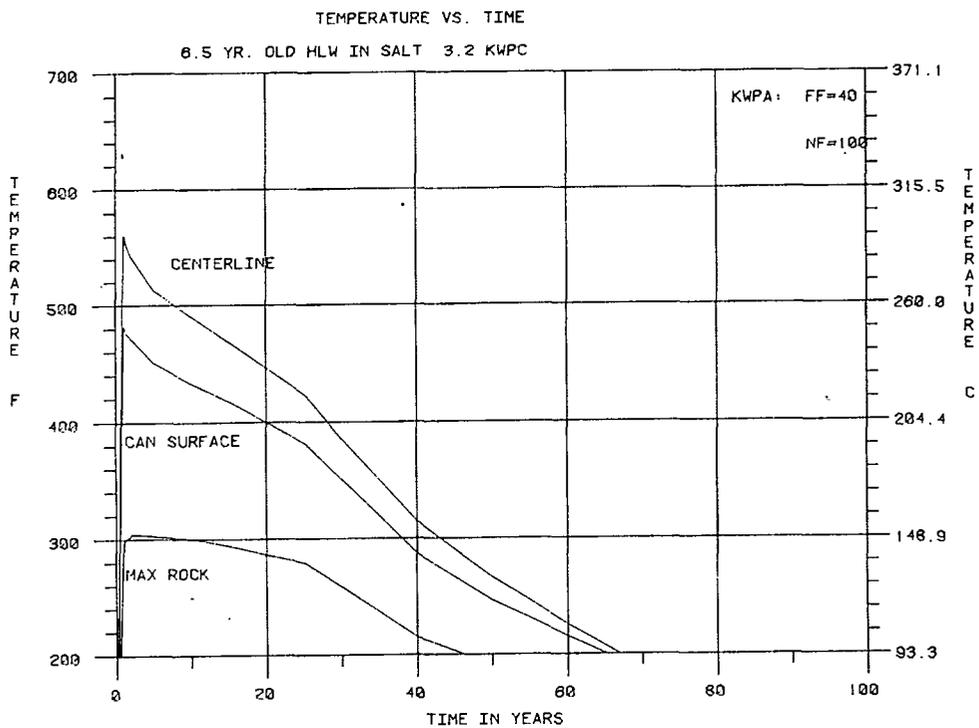


FIGURE K.1.10. Very-Near-Field Temperatures versus Time for Repository in Salt--Reprocessing Fuel Cycle

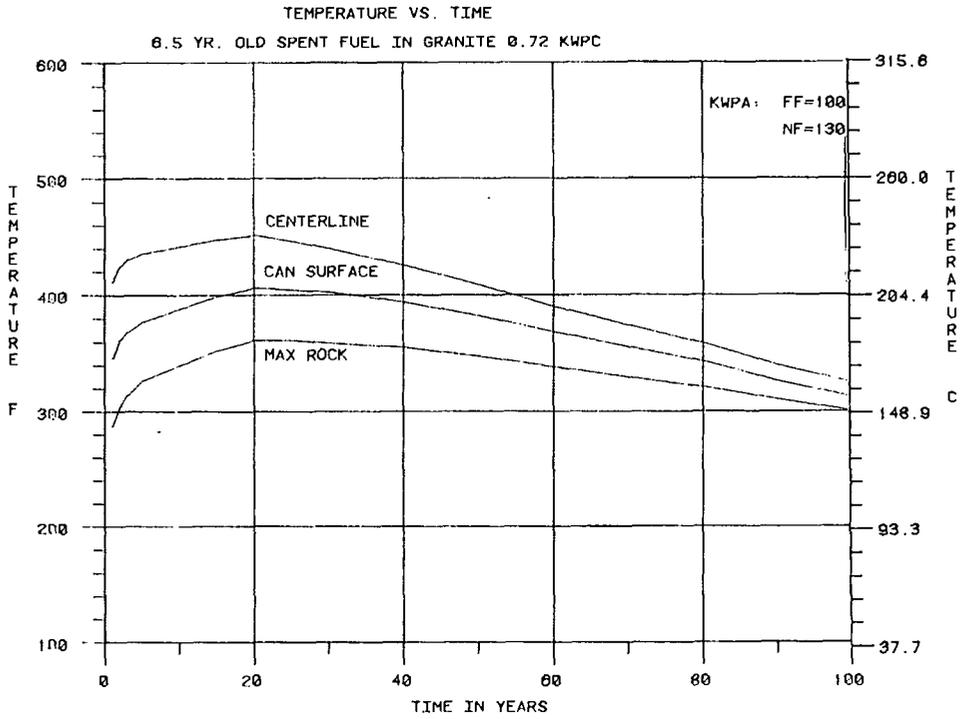


FIGURE K.1.11. Very-Near-Field Temperatures versus Time for Repository in Granite--Once-Through Fuel Cycle

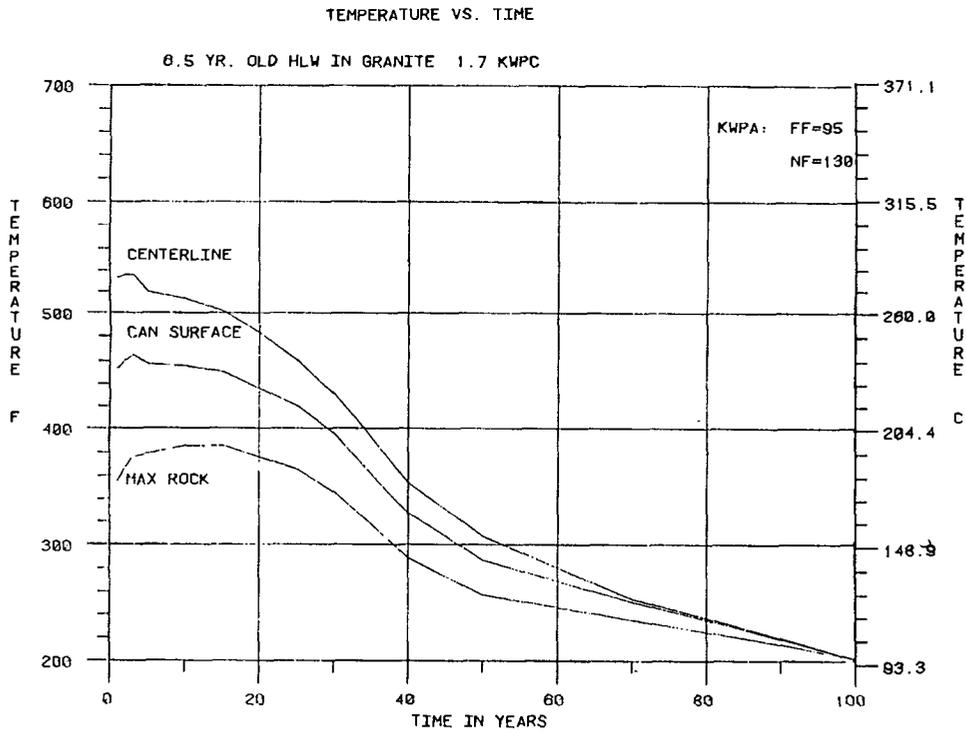


FIGURE K.1.12. Very-Near-Field Temperatures versus Time for Repository in Granite--Reprocessing Fuel Cycle

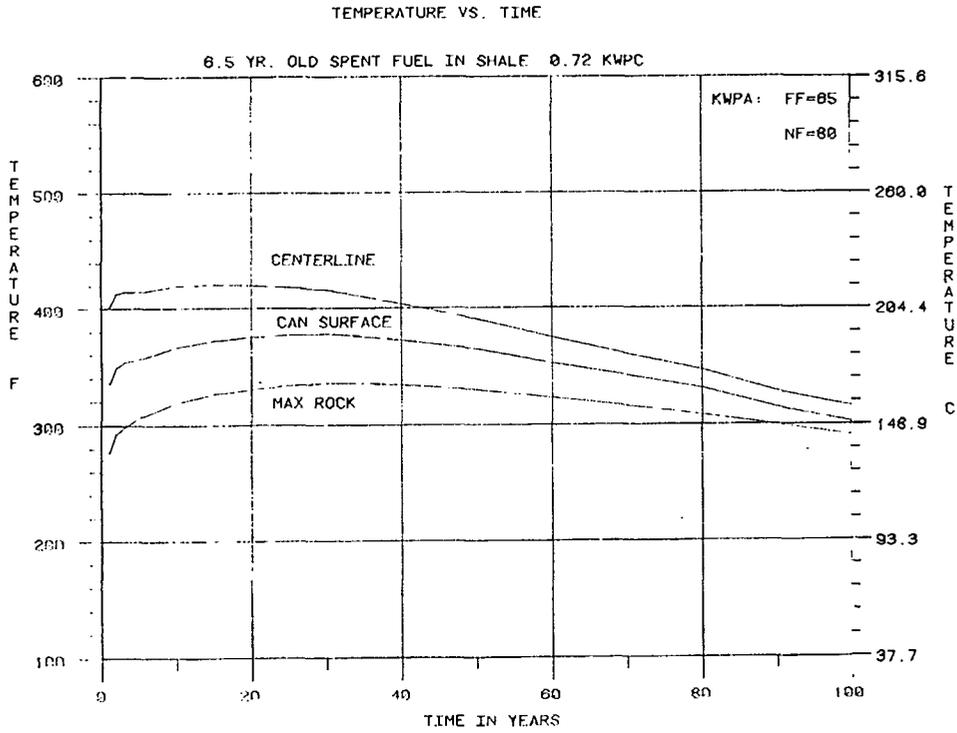


FIGURE K.1.13. Very-Near-Field Temperatures versus Time for Repository in Shale--Once-Through Fuel Cycle

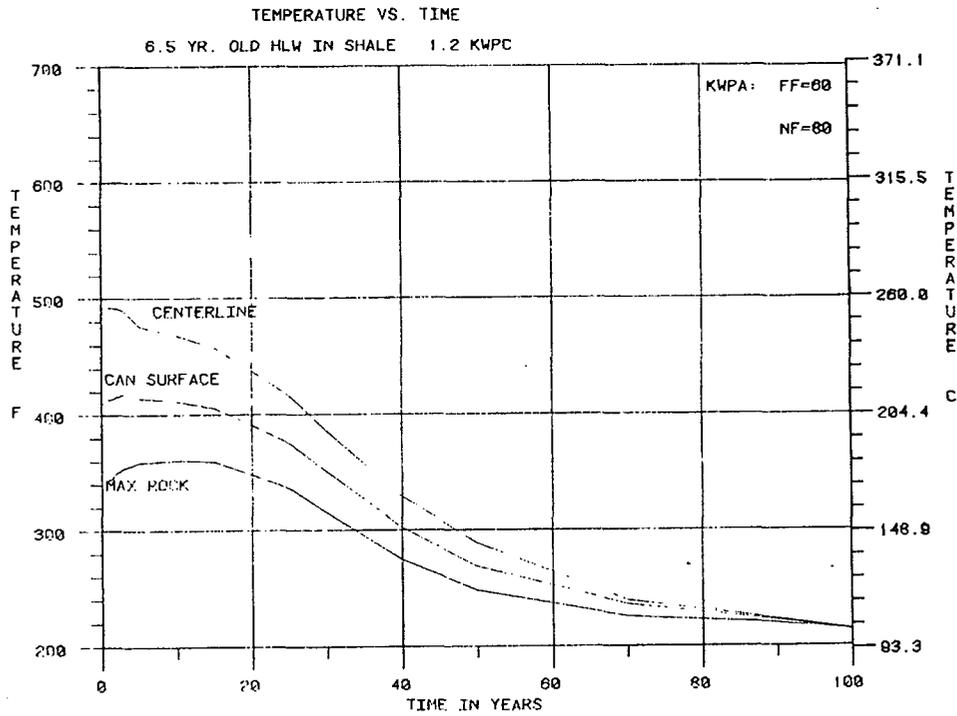


FIGURE K.1.14. Very-Near-Field Temperatures versus Time for Repository in Shale--Reprocessing Fuel Cycle

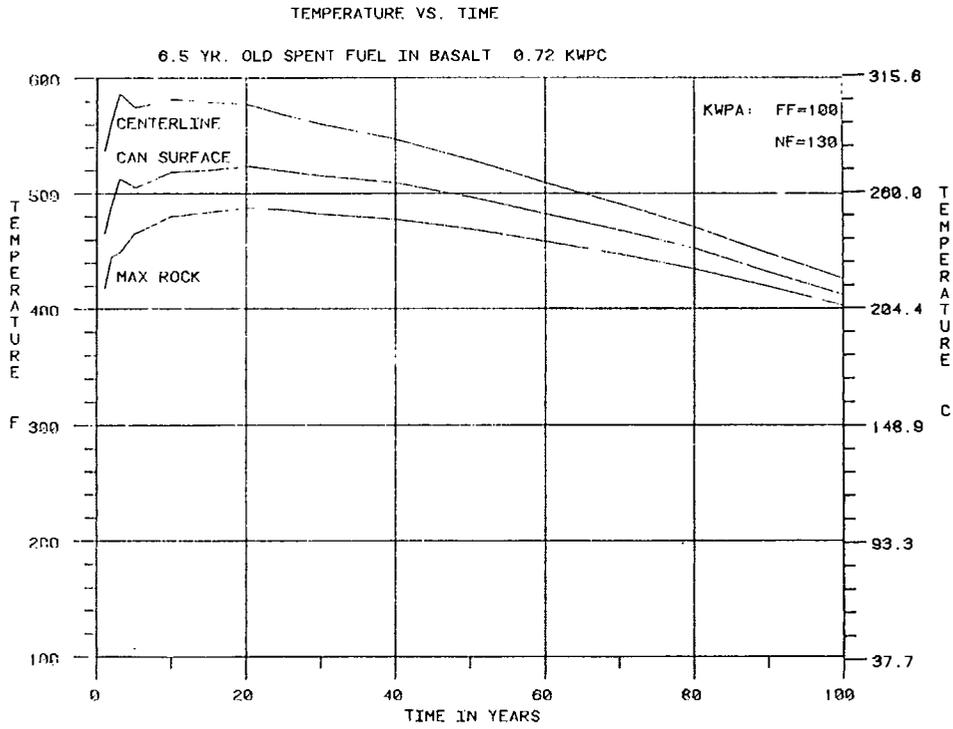


FIGURE K.1.15. Very-Near-Field Temperatures versus Time for Repository in Basalt--Once-Through Fuel Cycle

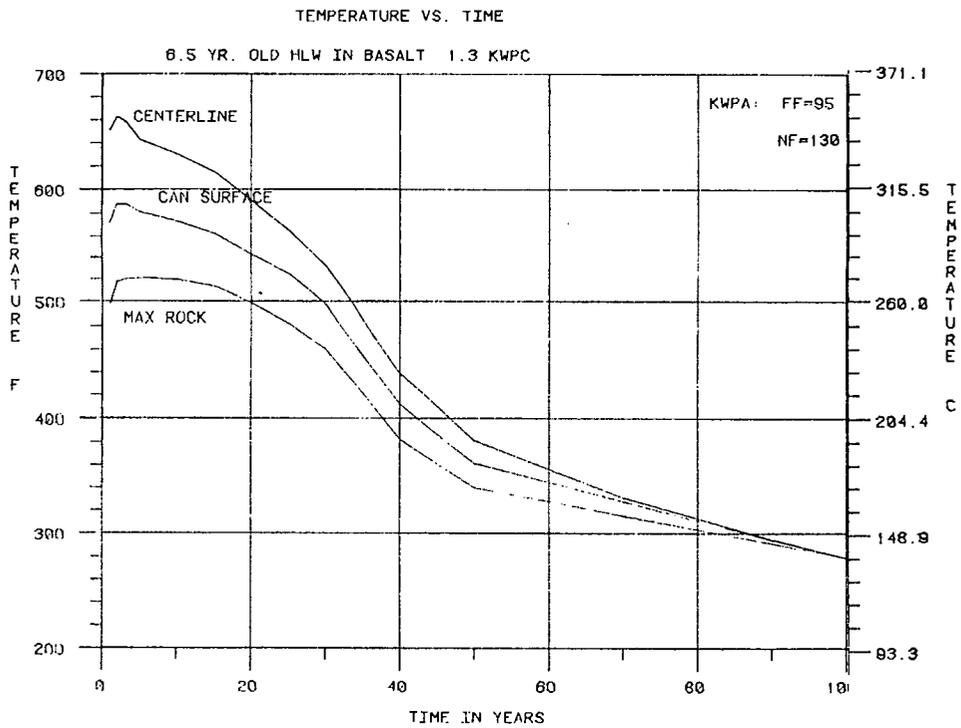


FIGURE K.1.16. Very-Near-Field Temperatures versus Time for Repository in Basalt--Reprocessing Fuel Cycle

TABLE K.1.5. Material Properties

	Density kg/m ³	Specific Heat W yr/kg °C	Thermal Conductivity W/m °C
Salt	2162.0	2.65×10^{-5}	(see Table K.1.6)
Shale	2563.2	2.65×10^{-5}	(see Table K.1.6)
Granite	2675.0	2.65×10^{-5}	(see Table K.1.6)
Basalt	2883.0	2.65×10^{-5}	(see Table K.1.6)
Waste	2995.7	2.65×10^{-5}	1.21
Concrete	2306.9	2.65×10^{-5}	0.935
Backfill	2563.2	2.65×10^{-5}	0.346

TABLE K.1.6. Thermal Conductivities W/m°C

Temperature (°C)	Basalt	Granite	Salt	Shale ^(a) (horizontal)
0	1.16	2.86	6.11	1.94
50	1.19	2.70	5.00	1.78
100	1.26	2.56	4.21	1.77
150	1.32	2.44	3.60	1.75
200	1.37	2.34	3.12	1.73
300	1.49	2.15	2.49	1.71
400	1.56	1.99	2.08	1.70

(a) Shale vertical conductivity = 0.739 x shale horizontal conductivity.

K.1.3 Impacts of Waste Age

The thermal criteria discussed in Section K.1.1 are calculated on the basis of 10-year-old waste. Criteria estimates for waste ages of 5 to 50 years were also developed. As spent fuel or HLW ages, the intensity of emitted radiation and heat declines and the quantity of these materials that can be emplaced in a given repository area increases somewhat. The thermal loading criteria required to meet the same temperature limits tend to decline for older wastes but heat emissions decline at a faster rate resulting overall in an increase in capacity.

Table K.1.7 lists maximum thermal loading criteria developed for both spent fuel and HLW at 5, 10, and 50 years of age. These loadings take into account the temperature and thermo-mechanical limitations listed in Table K.1.1.

The thermal loadings used to calculate repository capacities are shown in Table K.1.8. These loadings take into account: 1) loading at 2/3 of calculated maximum, 2) the relationship between the near-field and far-field areas (i.e., the unused passive areas for corridors, etc.), and 3) the limiting parameter, which is denoted by an asterisk.

TABLE K.1.7. Thermal Loading Limits for Waste Repositories (kW/Acre)

Formation	Age of Waste at Emplacement (yr)	Spent Fuel		HLW	
		Near-Field Local Areal Loading Limit	Far-Field Average Repository Loading Limit	Near-Field Local Areal Loading Limit	Far-Field Average Repository Loading Limit
Salt	5	240	100 ^(a)	190	190
	10	150	60 ^(a)	150	150
	50	100	31 ^(a)	130	80 ^(a)
Shale	5	180	180	140	140
	10	120	120	120	120
	50	70	63 ^(a)	120	120
Granite	5	300	300	210	210
	10	190	190	190	190
	50	140	140	180	180
Basalt	5	300	300	210	210
	10	190	190	190	190
	50	140	140	180	180

(a) Long-term far-field considerations limit average repository loading in these cases.

TABLE K.1.8. Thermal Loadings Used, kW/Acre

Formation	Age of Waste at Emplacement (yr)	Spent Fuel		HLW	
		Near-Field Local Loading	Far-Field Average Loading	Near-Field Local Loading	Far-Field Average Loading
Salt	5	84	67*	130*	97
	10	50	40*	100*	76
	50	25	20*	70	54*
Shale	5	120*	96	94*	70
	10	80*	65	80*	60
	50	52	42*	80*	60
Granite	5	200*	162	140*	108
	10	130*	105	130*	100
	50	94*	76	120*	93
Basalt	5	200*	162	140*	108
	10	130*	105	130*	100
	50	94*	76	120*	93

* Denotes limiting parameters.

The calculated repository capacities at these loadings are shown in Table K.1.9 assuming 2000-acre repositories. These results are plotted and discussed in Section 5.3. Maximum temperatures calculated for these loadings in both the near-field and far-field are shown in Tables K.1.10 through K.1.12. The thermal criteria are met in all cases except for spent fuel in basalt where spent fuel center pin temperature exceeds the 300°C criteria at both 5- and 10-year loadings, indicating that basalt capacities may be overstated. The variation in the maximum temperature in all media indicates that further optimization of the loading criteria is desirable.

TABLE K.1.9. Repository Capacities as a Function of Waste Age, MTHM

Waste Type and Median	5-Year Age	10-Year Age	50-Year Age
Spent Fuel			
Salt	57,600	61,100	64,700
Granite	141,000	150,000	193,000
Shale	70,700	76,300	90,600
Basalt	141,000	150,000	193,000
Reprocessing HLW			
Salt	66,300	83,200	124,000
Granite	66,200	89,700	137,000
Shale	36,900	46,300	68,000
Basalt	63,000	83,300	122,000

TABLE K.1.10. Maximum Near-Field Temperatures with Spent Fuel

Formation	Age of Waste	Maximum temperature, °C and Year		
		Formation	Canister	Waste Centerline
Salt	5	92 @ 30 yr	127 @ <1 yr	143 @ <1 yr
	10	88 @ 40 yr	106 @ 20 yr	116 @ 10 yr
	50	107 @ 80 yr	110 @ 80 yr	116 @ 80 yr
Granite	5	218 @ 30 yr	232 @ 20 yr	243 @ 1 yr
	10	227 @ 30 yr	238 @ 25 yr	243 @ 25 yr
	50	204 @ 70 yr	210 @ 70 yr	213 @ 70 yr
Shale	5	193 @ 30 yr	227 @ 1 yr	252 @ 1 yr
	10	204 @ 30 yr	221 @ 25 yr	227 @ 20 yr
	50	171 @ 60 yr	182 @ 40 yr	187 @ 40 yr
Basalt	5	288 @ 20 yr	312 @ 2 yr	332 @ 1 yr
	10	299 @ 30 yr	312 @ 30 yr	318 @ 25 yr
	50	254 @ 60 yr	260 @ 60 yr	262 @ 60 yr

TABLE K.1.11. Maximum Near-Field Temperatures with HLW

Formation	Age of Waste	Maximum temperature, °C and Year		
		Formation	Canister	Waste Centerline
Salt	5	160 @ 1 yr	334 @ <1 yr	416 @ <1 yr
	10	191 @ 10 yr	343 @ 3 yr	422 @ 2 yr
	50	160 @ 5 yr	332 @ <1 yr	415 @ <1 yr
Granite	5	180 @ 1 yr	279 @ <1 yr	321 @ <1 yr
	10	235 @ 15 yr	312 @ 10 yr	349 @ 3 yr
	50	242 @ 25 yr	306 @ 15 yr	344 @ 5 yr
Shale	5	179 @ 1 yr	243 @ 1 yr	266 @ 1 yr
	10	218 @ 10 yr	268 @ 10 yr	296 @ 10 yr
	50	232 @ 25 yr	277 @ 25 yr	302 @ 5 yr
Basalt	5	262 @ 1 yr	331 @ 1 yr	360 @ 1 yr
	10	318 @ 10 yr	374 @ 10 yr	402 @ 5 yr
	50	319 @ 25 yr	364 @ 25 yr	394 @ 5 yr

TABLE K.1.12. Maximum Far-Field Temperature Increases

Formation	Age of Waste	Spent Fuel		HLW	
		Max°C	Year	Max°C	Year
Salt	5	40	54	34	52
	10	38	54	58	34
	50	27	500	48	54
Granite	5	115	54	64	34
	10	120	86	87	54
	50	160	500	97	54
Shale	5	92	86	57	22
	10	102	100	73	34
	50	103	500	84	54
Basalt	5	144	86	87	22
	10	155	100	130	34
	50	175	500	125	34

The heat generation rates used in these calculations are shown for both spent fuel (PWR) and reprocessing HLW in Table K.1.13.

TABLE K.1.13. Heat Generation Rates for Spent Fuel and High-Level Wastes

Waste Age	Watts/MTHM	
	PWR Spent Fuel	HLW
5	2.18×10^3	1.90×10^3
10	1.18×10^3	1.08×10^3
20	9.45×10^2	8.0×10^2
30	7.7×10^3	6.0×10^2
40	6.5×10^2	4.5×10^2
50	5.6×10^2	3.6×10^2
100	3.0×10^2	1.2×10^2
200	1.6×10^2	2.5×10^1
300	1.3×10^2	1.8×10^1
400	1.1×10^2	1.6×10^1
500	9.5×10^1	1.4×10^1
*1000	5.5×10^1	7.5×10^0

K.2 REMOVAL OF EMPLACED WASTE

Once wastes are emplaced at a geologic repository it is considered unlikely that they will require removal. Confidence in the suitability of the repository to isolate wastes will be high at the time waste emplacement operations commence because of extensive preemplacement testing and exploration, thorough DOE and peer review, and NRC licensing. In spite of this, repository design takes into account the possible need to remove emplaced wastes. Conditions that may be postulated to require waste removal include:

- detection of defective canisters that require removal, repackaging, and reemplacement
- disqualification of a portion of the repository that necessitates removal and reemplacement of the affected canisters
- failure of in-situ tests and data acquired during monitoring of repository operations to provide sufficient confidence in long-term repository performance, which requires removal of wastes and abandonment of the repository site.

As discussed in Section 5.3.1.5, wastes are emplaced in a readily retrievable manner during initial operations and are emplaced recoverably during the remainder of repository operations. Removal of emplaced wastes will require different levels of effort depending upon the phase of repository operations during which removal takes place.

K.2.1 Readily Retrievable Emplacement

During the initial phase of repository operation wastes are emplaced so that they can be readily retrieved. During this period emplacement holes are lined with steel sleeves and sealed with removable concrete plugs. The sleeves and plugs ensure that the canisters are accessible and minimize corrosion or other damage. Verification of repository functions continues throughout the period of ready retrievability; extensive in-situ testing rock core analysis, and other confirmatory programs are performed. In-situ testing and monitoring include sensors for temperature, strain and pressure, and sampling systems for air and ground water installed with a statistically significant number of canisters. From these activities, additional data will become available for use in the various mechanistic, computational models that form the basis for long-term projections of performance.

Should a decision be made to extend the period of readily retrievable emplacement beyond the initial 5 years, the use of sleeve-lined holes and concrete plugs would continue and rooms would be left open. For extension beyond a few years, the areal thermal density of emplaced wastes may need to be decreased. By decreasing the amount of thermal energy stored in the rooms, thermal stresses in the ceiling and supporting pillars are reduced to the point where room opening stability can be reasonably assured for the longer period.

Table K.2.1 lists calculated near-field local thermal densities for 25-yr readily retrievable emplacement of 10-year-old spent fuel at the conceptual repositories located in salt, granite, shale, and basalt formations. Consistent with the conservative approach

TABLE K.2.1. Near Field Local Thermal Densities^(a) for 25-Year Ready Retrievalability of 10-year-old Spent Fuel

Near-Field Allowable Thermal Loading, kW/acre			
<u>Salt</u>	<u>Granite</u>	<u>Shale</u>	<u>Basalt</u>
24	53	36	53

(a) These densities are conservative values that are 2/3 of the calculated densities.

taken in the 5-yr readily retrievable case, the values in Table K.3.1 are two-thirds of the calculated maximum acceptable thermal densities for 25-yr ready retrievability.

As discussed previously in Section K.1, Thermal Criteria, the criteria controlling placement of spent fuel in salt with 5-yr ready retrievability is the far-field average thermal density. However, in the case of 25-yr ready retrievability, near-field local thermal density becomes the controlling criterion because maintaining room and pillar stability for 25 years requires a more restrictive thermal density than is needed to limit long-term uplift.

An additional concern for the repository in salt is the creep closure of rooms over the 25-yr period of ready retrievability. To compensate for this, room ceiling heights are increased 7.6 m for 25-yr ready retrievability (6.7 m for 5-yr ready retrievability).

An alternative approach for achieving 25-yr ready retrievability is to provide heat removal from the mine by continuously ventilating emplacement rooms. This technique could allow higher thermal densities by removing heat from the rock formation to keep room and pillar stresses within acceptable limits. Additional details of this approach are provided in Y/OWI/TM-44 (Union Carbide Corp. 1978).

The unit cost for providing 25-yr ready retrievability for emplaced spent fuel elements at a repository located in salt is \$90/kg HM (mid-1978 dollars) compared to \$54/kg HM for 5-yr ready retrievability. The primary reason for this difference in cost is the reduction of repository waste capacity by about a factor of two for the 25-yr ready retrievability option. Another contribution to the higher cost is \$70 million for additional mining and backfilling that is necessary as a result of increased ceiling height for the repository in salt. Use of sleeves for all emplaced wastes also costs an extra \$4 million annually. Unit costs for 25-yr readily retrievable emplacement of spent fuel in the other rock media would also increase although additional mining to increase ceiling height would not be required.

During the initial phase of readily retrievable emplacement, removal operations are relatively straight-forward. Because rooms are left open and the lined emplacement holes are sealed with removable concrete plugs, removal of emplaced wastes simply involves reversing the emplacement procedures. A transporter reenters the emplacement room and positions itself over the sealed hole. The concrete plug would be removed and the waste canister raised into the transporter. The transporter then delivers the canister to a waste receiving station where it is loaded into a shaft and lifted to the surface. On the surface the canisters are placed into temporary dry well storage until a new repository is ready.

K.2.2 Recoverable Emplacement

At the end of the period of readily retrievable emplacement (assumed to last 5 years for this conceptual repository), holes are no longer lined with sleeves or sealed with concrete plugs, and rooms are backfilled after being filled with waste. For the remainder of repository emplacement operations the wastes are considered to be recoverable with considerably greater effort than required for removal of readily retrievable wastes. Although sufficient confidence in repository performance existed to justify termination of readily retrievable emplacement, observations and measurements will likely continue.

Recovery operations are more complex after the phase of readily retrievable emplacement ends. Before removal operations could begin, backfilled rooms first have to be reexcavated. This is done using standard earth-moving equipment with care taken to avoid excessive damage to emplacement holes. Backfill is removed from emplacement holes using shielded boring equipment; again, care is taken to avoid damage to the hole or canister. At this point the waste canister is removed to the surface as described for the readily retrievable case.

In the event that a canister has become damaged and is not able to be extracted directly from the hole special steps need to be taken. This may include core drilling around the damaged canister through the surrounding rock. The rock and waste are then transported to the surface and repackaged for temporary storage and disposal elsewhere.

K.3 ENGINEERED SORPTION BARRIERS

In addition to retardation of radionuclide migration with an appropriate canister design or inert coating of the waste form, certain materials can be used to absorb or otherwise slow radionuclide migration from the package and the repository.

Possible retardation mechanisms include surface adsorption, ion exchange, coprecipitation, and redox effects. The use of coprecipitation appears impractical as a retardation mechanism because of its rather selective nature and because a wide range of radionuclide chemical species must be retarded.

K.3.1 Performance Requirements

Solids selected for radionuclide adsorption, ion exchange, and redox effects in several combinations can be used for repository backfilling, for an overpack in the immediate vicinity of the canister exterior, and/or for a protective packing between the waste form and the interior surface of the canister (Karn-Bransle-Sakerhet 1978). The sorption material must be mechanically, thermally, and chemically stable in the repository environment. Also, it must be dry when in contact with the canister interior and in the waste form radiation field to prevent accelerated canister corrosion or pressurization. Good heat conducting properties and relatively low permeability to ground water also are desirable sorption material characteristics. If the material is used for repository backfilling, it should have sufficient loadbearing capacity to prevent cavern roof collapse onto stored wastes and to prevent major movement of the waste canisters. The organic contents of the filling material should be very low, probably less than 1%, to avoid radionuclide complexing and enhanced migration rates. Materials may be added to affect oxidation-reduction changes that retard radionuclide migration. Radionuclide migration rates of the elements antimony, iodine, neptunium, plutonium, ruthenium, technetium, and uranium may be affected by changes in the redox potential.

K.3.2 Sorption Materials Performance

Research sponsored by the Office of Nuclear Waste Isolation (ONWI) is determining sorption coefficients of many minerals and rocks that may be of interest for sorption barrier use. Swedish (Allard et al. 1977, Haggblom 1977) and Canadian (Acres Consulting Services, Ltd. 1977) workers also have ongoing programs to investigate sorption of radionuclides in clays and rocks. Sorption investigations involving 19 radionuclides that are of interest in waste disposal operations were summarized in a 1976 EPA literature search (Ames and Rai 1978).

The solution species formed from the radionuclides of the various elements are a primary control on their adsorption by a potential retardant. Possible solution species, based on existing thermodynamic data, are shown in Table K.3.1. Rocks, soils, and sediments are predominately cation exchange materials. The inorganic anion exchangers include the amorphous hydrated oxides from iron, aluminum, and manganese, which are found naturally, and other synthetic anion exchange materials such as zirconia or titania. The environmental factors reported to effect radionuclide adsorption are summarized in Table K.3.2.

TABLE K.3.1. Predominant Solution Species of Elements Without Organic Ligands
(Karn-Bransle-Sakerhet 1978)

Elements	Little Affected by Oxidation-Reduction	In an Oxidizing Environment	In a Reducing Environment
Am	Am^{3+} , AmSO_4^+ , $\text{Am}(\text{OH})^{2+}$		
Sb		HSbO_2^0 , $\text{Sb}(\text{OH})_3^0$, SbOF^0 , $\text{Sb}(\text{OH})_4^-$	SbO^+
Ce	Ce^{3+} , CeSO_4^+		
Cs	Cs^+		
Co	Co^{2+} , $\text{Co}(\text{OH})_2^+$		
Cm	Cm^{3+} , CmOH^{2+} , $\text{Cm}(\text{OH})_2^+$		
Eu	Eu^{3+} , EuSO_4^+ , $\text{Eu}_2\text{P}_2\text{O}_7^{2+}$		
I		I^- , IO_3^-	I^-
Np		NpO_2^+ , $\text{NpO}_2\text{HPO}_4^-$, NpO_2HCO_3	NpOH^{3+} , Np^{4+}
Pu		PuO_2^{2+} , $\text{PuO}_2(\text{CO}_3)(\text{OH})_2^{2-}$, PuO_2^+	PuOH^{2+} , Pu^{3+}
Pm	Pm^{3+}		
Ra	Ra^{2+}		
Ru		$\text{Ru}(\text{OH})_2^{2+}$, RuO_4^- , RuO_4^{2-}	RuO_4^-
Sr	Sr^{2+}		
Tc		TcO_4^-	TcO_2
Th	ThF^{3+} , $\text{Th}(\text{OH})_3^+$		
^3H	H^+ , $^3\text{H-O-H}$		
U		UO_2^{2+} , UO_2F^+ , $\text{UO}_2(\text{OH}_2)^0$, $\text{UO}_2(\text{CO}_3)_3^{4-}$	UO_2^{2+} , UOH^{3+} , UO_2^+ , $\text{UO}_2(\text{CO}_3)_3^{4-}$
Zr	$\text{Zr}(\text{OH})_4^0$, $\text{Zr}(\text{OH})_5^-$, ZrF^{3+}		

Examples of inorganic sorption materials are given in Table K.3.3. Chabazite, erionite and clinoptilolite are zeolites that occur in large deposits of sedimentary origin (Hay 1966) and montmorillonite is the main clay mineral in bentonites. Thermal and hydrothermal stabilities generally are acceptable for the intended use. The thermal conductivities of both clay minerals and zeolites are comparatively low. The zeolites are quite permeable to ground-water while sodium-based montmorillonites show low permeabilities (Jacobsson 1977).

Pusch (1978) has suggested that varying amounts of quartz sand be added to bentonite and that it be compacted to improve its load-bearing and thermal conductive characteristics while retaining some of its cation exchange properties. Through the use of simple relationships between diffusion or solution flow-controlled migration and equilibrium distribution coefficients, Neretnieks (1977) determined the retention time in years in 1-m-10% bentonite/90% quartz and clinoptilolite sorption barriers as shown in Table K.3.4.

The barrier depths (in meters) required to retard various radionuclides for 30 half-lives are shown in Table K.3.5. Clinoptilolite is a better sorption barrier, but it is more permeable and has less bearing strength than the bentonite-quartz mixture. For use within the canister, the clay minerals and zeolites can be dehydrated at just below their stability temperatures.

TABLE K.3.2 Factors Reported to Effect Adsorption of Radioelements Over the pH Range of 4 to 9 (Karn-Bransle-Sakerhet 1978)

Element	Factors								Probable Adsorption Mechanisms
	pH	Eh	Soil CEC	Competing Ions	Selectively Adsorbed on	Inorganic Ligands	Complex Ions Organic Constituents	Colloid Formation	
Am			X	X		X			IE
Sb		X			Iron Oxides	X	X	X	PPT
Ce	X					X	X	X	TE, PPT
Cs			X		Zeolites, Micas				IE
Co	X				Illite, Iron Oxide	X	X	X	IE, PPT
Cm						X		X	PPT
Eu	X			X		X			IE, PPT
I					OM		X		OM
Np						X		X	UNK
Pu	X	X	X			X	X	X	IE, PPT
Pm	X			X			X		IE, PPT
Ra			X	X	Zeolites, Barite				IE
Ru	X				OM	X	X		PPT
Sr	X		X	X	Calcite, Zeolites	X	X		IE
Tc							X		UNK
Th	X		X		OM	X	X	X	IE, PPT
³ H					H ₂ O				NONE
U	X	X	X		OM	X	X		PPT, IE
Zr	X					X	X	X	PPT

(a) CEC = Cation Exchange Capacity.

(b) IE = Ion exchange, OM = Organic Matter Adsorption, PPT = Precipitation, UNK = Unknown.

TABLE K.3.3. Thermal Stabilities and Cation Exchange Capacities of Several Clay Minerals and Zeolites (Breck 1974, Eberl et al. 1978, Ames and Sand 1958, Grim 1968)

Material	Composition	Hydrated Cation Exchange Capacity, meq/100 g	Thermally Unstable in Air at, °C
Chabazite	Ca ₂ [(AlO ₂) ₄ (SiO ₂) ₈] · 3H ₂ O	390	600
Erionite	Ca _{4.5} [(AlO ₂) ₉ (SiO ₂) ₂₇] · 27H ₂ O	310	750
Clinoptilolite	Ca ₃ [(AlO ₂) ₆ (SiO ₂) ₃₀] · 24H ₂ O	220	750
Mordenite	Ca ₄ [(AlO ₂) ₈ (SiO ₂) ₄₀] · 24H ₂ O	230	800
Montmorillonite	(Al _{3.34} Mg _{0.66})Si ₈ O ₂₀ (OH) ₄ · H ₂ O	150	390

Hydrothermal Stabilities at 100 Bars Pressure for 10 Days

	°C	Products	Composition
Chabazite	230	Wairakite	Ca ₈ (AlO ₂) ₁₆ (SiO ₂) ₃₂ · 16H ₂ O
Clinoptilolite	360	Mordenite	
Montmorillonite	400	Quartz, Feldspar	SiO ₂ , NaAl ₃ Si ₃ O ₈
Montmorillonite	300	Quartz, Montmorillonite	

TABLE K.3.4 Retention Time Ranges on 1-m Barriers for Several Radionuclides (Neretnieks 1977)

Radionuclide	Retention Time, Years	
	10% Bentonite/ 90% Quartz	Clinoptilolite
^{90}Sr	30	600 to 1,400
^{137}Cs	20 to 30	2,200 to 5,200
^{226}Ra	40 to 50	600 to 1,400
^{229}Th	50 to 300	unknown
^{237}Np	2.1×10^6	unknown
^{239}Pu	2.4×10^4	unknown
^{241}Am	458	1,000 to 30,000
^{99}Tc	1	1
^{129}I	1	1

TABLE K.3.5 Barrier Depth (m) Required to Retard Various Radionuclides 30 Half-Lives (Neretnieks 1977)

Radionuclide	30 Half-Lives Barrier Depth, m	
	10% Bentonite/ 90% Quartz	Clinoptilolite
^{90}Sr	1	0.2
^{137}Cs	1	0.1
^{226}Ra	40	1.5
^{241}Am	1	0.1 to 0.7

If the reduced or oxidized species is less soluble than the original radionuclide solution species, an oxidation-reduction (redox) reaction may be used to retard the mobilities of certain radionuclides. Very little work has been done using redox controlling materials as migration retardants. An example of redox control is the use of wustite (FeO) to surround the waste form. The oxidation of ferrous to ferric ion would reduce technetium in the highly mobile pertechnetate ion (TcO_4^-) from Tc(VII) to Tc(IV). (Latimer 1952 and Pourbaix 1966). Tc(IV) is a much less mobile form of technetium than Tc(VII).

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APPENDIX L

WASTE ISOLATION RESEARCH AND DEVELOPMENT PROGRAM

A program of waste isolation research and development is underway to obtain the data identified as needed in this report, as well as those identified in other review activities. The Department of Energy (DOE), conducting research and development toward waste isolation, has placed emphasis on the development of plans and strategies which incorporate an iterative approach which includes substantial scientific peer review. An important activity of this type was the preparation of the Earth Science Technical Plan (ESTP) for Disposal of Radioactive Waste in a Mined Repository (DOE/USGS 1980). The ESTP describes the research and development programs sponsored by the Departments of Energy and Interior. Additional work is in progress in the U.S. sponsored by the Nuclear Regulatory Commission (NRC), Environmental Protection Agency (EPA), and the utility industry. Additional work is also in progress in Sweden, Germany, France, England, Japan and Russia. A list of ongoing research projects organized by the technology categorization of Section 5.2 (Geologic Disposal--States of Technology and Research and Development) is presented below. This list is not complete, but rather is intended to suggest the scope and depth of current research.

L.1 GEOLOGIC SITE SELECTION

Research and development projects supporting site selection technology are listed below by several subcategories.

L.1.1 Long Term Geologic Stability

ONWI ^(a)	Regional studies to exclude tectonically active areas from further siting considerations
LASL/SLA	Prediction of volcanic activity
LLL/SLA	Flow charts for investigation and evaluation of candidate sites
LLL	Derivation of parameters for evaluating sites
ONWI	Criteria for geologic disposal of radioactive waste and site qualification criteria
DOE/Woodward-Clyde	Evaluation of the Paradox Basin
DOE/TBEG	Evaluation of the West Texas Bedded Salt
DOE/Law Eng.	Evaluation of Gulf Coast Salt Domes
DOE/Stone & Webster	Evaluation of the Salina Salt Basin

(a) The research and development organizations indicated by abbreviation are identified on page L.1.7.

L.2

USGS/LASL-SLA	Evaluation of the Nevada Test Site and Southern Nevada
DOE/RHO et al.	Evaluation of the Columbia Plateau
DOE/PNL	Release scenario modeling
DOE/LLL	Derivation of parameters for evaluating sites
ONWI	Criteria for geologic disposal of radioactive waste and site qualification criteria
USGS	Long term prediction of natural events and changes
DOE/SLA	Climatic/tectonic stability of the West Texas Salt Flats Basin
USGS	Climatic stability, Pecos River history
DOE/SLA	Climatic stability, San Simon Sink.

L.1.2 Characterization of Current Geology and Hydrology

USGS	Radar techniques, high-frequency electromagnetic borehole techniques, geophysics for site characterization
LLL/Texas A&M	Radar exploration techniques
Texas Bur. Mines/CONOCO	Improving resolution of existing geophysical techniques
DOE/LBL	Evaluation of geophysical techniques in fractured crystalline rock
Georgia Tech.	Geothermometry
DOE/ORO	The utility of petroleum exploration data in delineating structural features within salt anticlines
USGS	Water flux in the unsaturated zone of deserts, field test of flow in unsaturated alluvium, nonisothermal water fluxes in the unsaturated zone, characterization of local ground water systems, short-term hydraulic effects, fluid flow in fractured rocks, solute transport model in the unsaturated zone
USGS/SLA	Characterization of regional ground water systems
DOE/LLL	Fracture permeability of rocks under pressure, permeability measurements
DOE/LBL	Crystalline rock hydrology.

L.1.3 Seismicity

DOE/SRL	Subsurface earthquake damage
DOE/SLA	Effect of depth on ground motion

L.3

NRC/NUREG	Workshop/review of site suitability criteria
ONWI	Geological criteria for suitable sites of high-level radioactive waste; criteria for the geologic disposal of radioactive waste and site qualification criteria, preliminary site selection for SPR facilities in Louisiana and Mississippi .
OWI/Woodward-Clyde	Preliminary geologic site-selection criteria for NWTS.

L.1.4 Land Use and Transportation Considerations

TRW	Socioeconomic studies
HARC	Socioeconomic studies
Stearns-Roger	Conceptual Design No. 1
Kaiser	Conceptual Design No. 2; SAI
NUS	Environmental Criteria.

L.2 HOST ROCK PROPERTIES

Research and development projects to better define host rock properties are listed below by several subcategories.

L.2.1 Discontinuities

USGS	Development of geophysical techniques, high frequency electromagnetic borehole techniques
LLL	Development of single hole electromagnetic probe
RHO	Verification studies of specific geologic structures of the Columbia Plateau.

L.2.2 Rock Strength and Excavation Stability

USGS/LASL-SLA	Evaluation of granite, argillite, and tuff at the Nevada Test Site and in southern Nevada
USGS	Surveys of granite and other crystalline rocks, argillaceous rocks, western Cretaceous shales tuff and zeolitized tuff
DOE/LBL	Directional permeability of Stripa granite
DOE/BNL	Geothermometry of shale
DOE/SLA	In-situ test of Conasauga Shale
DOE/LLL	Granite heater and rock mechanics test Climax Stock.

L.2.3 Hardness and Mineability

ORNL/SAI	Expected repository environments
RE/SPEC	Repository concepts analysis
Univ. of Minn.	Development of displacement-discontinuity models
ORNL	Salt model pillar studies
USGS	Geomechanics
RHO/CSM	Advanced rock testing of basalt
LLL	Mechanical behavior of rocks under pressure
LBL	Material behavior of strips granite
LBL	Ultra-large rock core tests
RHO/PNL/and Others	Field investigation to determine in-situ geologic, hydrologic, and engineering parameters.

L.2.4 Rock Permeability and Ground Water Flow

LBL	Development of fractured flow and thermal-hydraulic flow models
USGS	Solution of solute transport equations
LBL	Development of analytical transport models
UCB	Brine migration modeling
SLA	Tracer tests of overlying strata
SRL	Osmotic effects of clay minerals
USGS	Solute transport in the unsaturated zone
USGS	Water flux in the unsaturated zones of deserts
USGS	Field test of flow in unsaturated alluvium.

L.2.5 Rock Pressure

DOE/LBL	In-situ stress measurements techniques (Stripa 7). In-situ thermomechanical test in Stripa granite
DOE/RE/SPEC	In-situ test--Avery Island Salt Dome
DOE/SLA	Instrumentation development for in-situ tests. Thermal-structural interaction--bench and in-situ tests. In-situ test of Conasauga Shale
DOE/LLL	Granite heater and rock mechanics tests
DOE/RHO	Near-surface Test Facility programs for in-situ testing of basalt at the Hanford Reservation, rock mechanics methods and in-situ heater tests in basalt.

L.3 THERMAL AND RADIATION EFFECTS

DOE/LLL	Mechanical behavior of rocks under pressure
DOE/SDSM	Bench-scale creep tests on rock salt
DOE/Texas A&M	Transient creep in rock salt
DOE/SLA	Thermal-structural interactions
DOE/SAI-LBL	Analysis of thermomechanical response of salt
USGS	Geomechanics of thermally induced stress on in-situ stress and fracturing
DOE/LBL	In-situ stress measurement techniques (Stripa 7)
DOE/RHO-Univ. of Minn. & Dames & Moore	Numerical modeling of rock stresses within a basaltic nuclear repository
NRC/TASC	Information base for waste repository design, Volume 3; Waste Rock Interactions
DOE/ORNL	Radiolysis of brine
DOE/SLA	Thermal-structural interactions in salt, pressure effects on thermal conductivity and expansion of geologic materials
DOE/LBL	In-situ thermomechanical tests of Stripa granite, large scale permeability tests of granite in the Stripa Mine and thermal conductivity tests
DOE/LLL	Granite heater and rock mechanics tests Climax Stock
DOE/RE/SPEC	Parametric thermoelastic analysis of high-level waste and spent fuel repositories in granite and other non-salt rock types.

L.4 REPOSITORY PERFORMANCE

DOE/ORNL	Borehole plugging--cement technology studies
DOE/SLA	Materials development, instrumentation, and field testing for borehole plugging
NRC/TASC	Information base for waste repository design, Volume 1, Borehole and Shaft Sealing
DOE/RHO	Borehole plugging programs at Hanford
ONWI/PNL	Borehole plugging and shaft sealing for geologic isolation of radioactive waste.
ONWI/PNL	Assessment of the effectiveness of geologic isolation systems
ONWI/PNL	Waste/rock interaction technology
NRC/SLA	Risk methodology for radioactive waste disposal in geologic media
NRC/LLL	Standards for the management and disposal of high-level and transuranic waste
EPA/Univ. of N.M.	Assessment method for geologic isolation of nuclear waste
EPA/ADL	Technical base for establishing regulations for disposal of HLW
USGS	Long term prediction of natural events and change
RHO/Furgo, Inc.	Assessment of geothermal and volcanic activity
SLA/LASL	Evaluation of tectonic, seismic, and volcanic hazards, Nevada Test Site and vicinity
BDM/INTERA/SAI/SLA	Nuclear waste repository safety assessment

Glossary of Acronyms Used in Appendix L

ADL	Arthur D. Little, Inc.
BDM	BDM Corporation
BNL	Brookhaven National Laboratory
BWIP	Basalt Waste Isolation Program
CONOCO	Continental Oil Co.
CSM	Colorado School of Mines
DEIS	Draft Environmental Impact Statement
DOE	U.S. Department of Energy
EIS	Environmental Impact Statement
EPA	U.S. Environmental Protection Agency
GCR	Geologic Characterization Report (WIPPO)
HARC	Human Affairs Research Center (Battelle)
HLW	High-Level Waste
INTERA	INTERA Environmental Consultants
LASL	Los Alamos Scientific Laboratory
LBL	Lawrence Berkely Laboratory
LLL	Lawrence Livermore Laboratory
N	Subcontractor not determined
NRC	U.S. Nuclear Regulatory Commission
NWTS	National Waste Terminal Storage
NUS	NUS Corp
ONWI	Office of Nuclear Waste Isolation (Battelle)
ORNL	Oak Ridge National Laboratory
ORO	Oak Ridge Operations (DOE)
OWI	Office of Waste Isolation
PBQ&D	Parsons, Brinkerhoff, Quade & Douglas
PIR	Preliminary Information Report
PNL	Pacific Northwest Laboratory
RE/SPEC	Re/Spec, Inc.
RHO	Rockwell Hanford Operation
SAI	Science Applications, Inc.
SDSM	South Dakota School of Mines
SLA	Sandia Laboratories-Albuquerque
SRL	Savannah River Laboratory
SSA	Southern Science Applications, Inc.
TASC	The Analytic Sciences Corporation
TBEG	Texas Bureau of Economic Geology
TRW	TRW Inc.
USGS	U.S. Geologic Survey
WIPP	Waste Isolation Pilot Plant
WISAP	Waste Isolation Safety Assessment Program

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APPENDIX N

WASTES FROM THORIUM-BASED FUEL CYCLE ALTERNATIVES

A number of thorium-based fuel cycle alternatives to the light water uranium-plutonium cycle have been proposed. The alternatives include: the uniform enrichment of thorium in LWR and heavy water; spike blanket systems in LWRs; crossed progeny in LWR's heavy water and fast converters; light water breeder (LWBR); and high-temperature gas-cooled reactor (HTGR) fuel cycles. For this Statement the LWBR and HTGR cycles have been chosen for discussion because their demonstration is nearer completion. Thus, they may be the first systems able to employ a thorium load. Moreover, a standard LWR using a thorium fuel cycle will have fission-product yields very similar to those of the LWBR. Analyses for managing wastes from these thorium fuel cycles have not been made in as great detail as for the LWR uranium cycles presented elsewhere in this Statement. The basis for this discussion is DOE/ET-0028, and that document should be referred to for a more-complete presentation.

As in the slightly enriched light water reactor cycle, power reactors could use thorium in either recycle or nonrecycle modes. In the recycle mode, spent fuel is reprocessed to remove fissionable ^{233}U that has been generated and to remove the initial fissionable species that remains unburned from the irradiation. (a) This material (mostly bred ^{233}U) can then be refabricated into fuel elements for reinsertion into a nuclear power reactor. This can be accomplished whether or not the amount of fissile material generated is large enough for the reactor to constitute a true breeder, which, once started, provides its own fissionable fuel. The system may not be operated as a breeder, but even so, the fissionable material required for makeup (^{235}U , plutonium, ^{233}U from other sources) may not be large.

In the nonrecycle mode, the fissionable material generated is not returned to the core, either because the fuel is not reprocessed or because the product from the reprocessing plant is treated as waste or is stored for future use. In this case, new fissionable material would be supplied for each core loading.

In the discussion that follows, wastes from the reprocessing of thorium fuels from LWBR and HTGR are compared with those from commercial light waste reactors (LWRs). It is assumed that 99.5% of the plutonium is separated from the LWR waste in reprocessing, but is not recycled. All comparisons are based on production of equal quantities of electrical energy.

(a) Under DOE management directives it is mandatory that ^{233}U and ^{239}Pu be disposed of in a similar manner. The reasoning for this is not because of any near-term risk from the ^{233}U but because of the higher specific toxicity of the daughter products in the long term.

Fission product activity in thorium wastes is about the same as that in LWR wastes, with only slight aggregate differences because of the mass distribution of ^{233}U fission fragments and the greater thermal efficiency of HTGRs. Some of the specific isotope yields are different by a factor of about two, but these differences are not among controlling long-lived isotopes and thus neither simplify nor complicate long-term waste storage as visualized and being developed for the slightly enriched uranium (SEU) cycle in LWRs.

Radiogenic heating is of importance when considering storage and isolation of certain radioactive wastes. Heat generation rates in the thorium wastes are essentially the same as in LWR wastes for the first several thousand years. They reach a maximum at less than twice the LWR rate in about 100 years, then decrease and finally peak again at 50 to 100 thousand years. Although the latter peak can exceed the LWR rate by a factor of 15, the actual value of the heat generation rate is quite small by that time.

For the first few thousand years, actinide and heavy element radioactivity in LWBR wastes is somewhat less than that in the LWR wastes. The radioactivity in HTGR wastes at these times exceeds that in LWR wastes by up to a factor of 7, because of the plutonium (primarily ^{238}Pu) which is present. After hundreds of thousands of years, the radioactivity in both HTGR and LWBR wastes exceeds that of LWR wastes by factors of 10 to 20. As in the case of heat generation, however, the absolute activity at these long times is relatively small.

In the instance of thermal neutron reactors, the more ^{233}U recycled, the lower will be the releases of transuranium isotopes formed by successive neutron captures in the fuel. This is due mainly to the fact that the capture-to-fission ratio is less for ^{233}U than for ^{235}U , ^{239}Pu , or ^{241}Pu . On the other hand, more (5 to 10%) ^{233}U in thorium fuel cycles must be fissioned than ^{235}U or plutonium in the SEU because the energy yield per fission for ^{233}U is less, and because thorium has about one-fifth the fast-fission effect of ^{238}U .

The actinide radioactivity and the heat generation rate differences are also influenced by the way the transuranic isotopes are managed, in particular regarding the yields on processing and the goal exposure of the fuels. However, when the gross characteristics of the LWBR-generated waste (total activity, heat output, chemical and physical form) are compared to LWR-generated waste, these characteristics are very similar (DOE 1979). As a result, no special waste management requirements are posed by the LWBR concepts which do not already exist for the LWR and no changes are anticipated to be necessary in the waste isolation program for LWR systems to accommodate a thorium-based system. ERDA (1976) performs an environmental assessment of a thorium-uranium fuel cycle and should be referred to for detailed information.

Gaseous releases from a facility reprocessing thorium- ^{233}U fuel would be somewhat greater than those from a reprocessing facility. This is particularly true for ^{85}Kr , although the xenon yields are more nearly equal. Because of the greater ^{85}Kr release, an analysis is required to determine the significance of the release.

The ^{14}C release from an HTGR reprocessing facility could be up to 15 times larger than that of SEU in LWRs because of the large amount of graphite in the fuel and the burning

operation used to separate the fuel from the structural material. In this amount a system might be required to remove ^{14}C from the reprocessing facility off-gas stream.

Because of high radioactivities, the isotopes ^{232}U and ^{228}Th incidentally generated in the ^{232}U -thorium cycle pose some short-term handling problems not significantly present with the SEU system. Uranium-232 has a 70-year half-life followed by much shorter half-lived daughters leading to ^{208}Tl , which has a 2.6 MeV gamma. In the recycle case, this complicates the handling of ^{233}U fuels even though it is present in only a few parts per million. However, as a diluent in uranium, it does not appreciably complicate waste storage. A long-term concern may be the precursor of ^{232}U , namely ^{231}Pa . The concentration in the wastes of ^{231}Pa with its 32,500-year half-life will depend on how it is managed in the successive recycle. There is, of course, an incentive to hold the protactinium in a processing vessel to assure that the ^{233}Pa fully decays to ^{233}U , which is then bled off and recycled. Under these circumstances there is no reason to recycle protactinium and thus ^{231}Pa is not "burned out." Its concentration in the wastes is correspondingly increased to levels that may approach the ^{239}Pu concentration in wastes from plutonium recycle. This could be alleviated by purposefully irradiating ^{231}Pa as an isolated target and by adding the ^{232}U generated into the high-level wastes in dilutions so localized heating will not be produced.

Currently it is believed necessary to add fluorine to dissolve spent thorium oxide fuel. The effects of fluorine, if any, upon the waste processing are unknown. However, steps could be taken to obviate the fluorine in the processing. This may involve addition of magnesium, calcium, or other elements to thorium oxide which will add to waste volume, but not appreciably to radioactivity. This may, however, increase the solubility of thorium dioxide in water coolant streams, increasing contamination of water coolant streams if fuel jackets develop leaks.

It is not believed that fluorine will detract from the qualities of the waste glass as fluoride is a constituent of many commercial glasses and enamels. The fluorine content of commercial glasses rarely exceeds 6%. Fluoride at those high concentrations acts as an opacifier in the glass owing to dispersed fluoride crystals. considerable laboratory experimentation has already been done on the incorporation of fluoride in nuclear waste glass.

REFERENCES FOR APPENDIX N

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APPENDIX P

MINERALS THAT COULD BE USED TO CONTAIN RADIONUCLIDES

This appendix presents a review of minerals that are candidates for the isolation of radionuclides in synthetic minerals, as discussed in Section 4.3.2.3. Analyses of the potential hazard from certain HLLW radionuclides suggest the greatest effort in solidification into synthetic minerals would be placed on the following groups of elements:

- Actinides and Lanthanides. The actinides Np, Pu, Am, Cm and their daughters constitute the major hazard in nuclear wastes from about 1000 to 5000 years of storage, with the exception of ^{226}Ra , which does not become significant until about 10^5 years (Cohen 1977). The majority of the lanthanide elements (La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Dy, Ho) are present as stable isotopes after a few years, and only trace amounts of a few Sm and Eu radionuclides have long half-lives. However, the lanthanides could be included with the actinides for several reasons: they occur together at one stage of partitioning; lanthanides and actinides are crystallographically and chemically very similar and usually occur together in the same minerals; the lanthanides can act as diluents in synthetic minerals for α -emitting actinides in order to minimize radiation effects.
- Strontium and cesium. These elements constitute both the major heat producers and bio-hazards in nuclear wastes for the first 600 years or so (Cohen 1977).
- Technetium and iodine. These two fission products have long-lived isotopes (^{99}Tc , $t_{1/2} = 2.1 \times 10^5$ y; ^{129}I , $t_{1/2} = 1.7 \times 10^7$ y) and are biohazards. They have the additional characteristics of forming anions that can migrate in soils and rocks as fast as the solutions in which they are dissolved (Rai and Seine 1978), i.e., without any substantial hold-up due to ion exchange or adsorption.

The minerals reviewed here are either known to contain substantial amounts of these elements or are likely to accept these elements based on compatible crystal chemistry. The physico-chemical and crystal chemical criteria for selecting host minerals, along with the common mineral synthesis methods, are discussed and tables of candidates are presented. A thorough treatment of what is known about the process of metamictization and metamict minerals is also included.

P.1 PHYSICO-CHEMICAL PRINCIPLESP.1.1 Stability Criteria

The physical and chemical foundations used to define whether a known mineral is classified as very stable, relatively unstable, or very unstable with respect to alteration, weathering and diagenesis include solubility and geologic data.

P.1.1.1 Use of Solubility Data

Chemical weathering and alteration are most often the result of the interaction between an electrolyte aqueous solution and the various minerals being weathered. Several factors are important in determining the mobility of elements via weathering ionic solutions. One group of factors is related to the overall physical properties of the "weathering system," i.e., of the hydrologic system and the host mineral assemblage. For example, the flow rate of solution through a permeable system is determined by Darcy's Law:

$$\vec{u} = -\frac{k}{\mu} (\rho \vec{g} + \nabla P) \quad v = \frac{\vec{u}}{\rho \emptyset}$$

where \vec{u} = fluid flux vector (g/cm²/sec)
 \vec{v} = true fluid velocity (cm/sec)
 \vec{g} = gravity force vector (cm/sec²)
 k = permeability of the rock assemblage (cm²)
 μ = viscosity of the fluid (cm²/sec)
 ρ = density of the fluid (g/cm³)
 P = pressure (bars)
 \emptyset = porosity of rock.

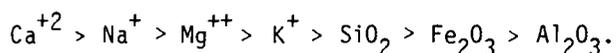
Clearly, then, the water flow depends on gravity and the pressure gradient at the given locality (a property of the hydrologic system as a whole) as well as on the porosity and permeability of the rock assemblage in the locality, and the density and viscosity of the fluid.

The hydrodynamic equations, which incorporate Darcy's Law, allow us to calculate the hydrodynamic mobility of a given cation or anion in solution from its original location within a given mineral of the weathered rock to its place of deposition such as a sedimentary deposit, rivers, oceans or the biosphere. We can obtain absolute flux rates for a given ion (i.e., moles/cm²/sec), however; only if we know its concentration in the percolating solution.

The magnitude of the concentration of a given element in a solution that is in contact with a weathering mineral assemblage is the central element used in establishing the intrinsic stability of a particular nuclear waste element-containing mineral to alteration and weathering. This concentration is generally a function of time, since it is kinetically controlled. Nevertheless, almost all geochemical work on the mobility of elements via solutions has applied a thermodynamic and not a true kinetic approach. Whether true thermodynamic equilibrium is reached between solution and a particular mineral depends, among other things, on how long they are in contact (i.e., the flow rate); this concept often appears as the ambiguous "water-rock ratio" in the literature. It seems likely that under most circumstances the concentration of an element in a weathering solution will be kinetically controlled. Unfortunately, there is a dire need for suitable kinetic data. The kinetic factors involved in the time dependence of the concentration, which may keep the concentration well below the thermodynamic limit, will be discussed below.

One can usually establish only an upper limit to the concentration of a given element by the use of thermodynamics. Assuming equilibrium between minerals and solution, the concentration of any particular nuclear waste element will then be governed by the solubility of the minerals containing it.

Before discussing the thermodynamic approach to stability, a brief review of the general qualitative work on weathering stability in the literature is presented. Soil geochemists have set up a qualitative scale of the different inherent tendencies of minerals to alter by weathering processes. The weathering rate depends on the structure and composition of the minerals, as well as the weathering environment. Goldich (1938) formulated such a weathering stability series for the major elements. He found that the major elements are removed from rocks and minerals in the order:



Loughnan (1969) gives a similar result (see Table P.1.1).

Much less is known about the relative mobilities of the trace elements (lanthanides, actinides, and others). Jackson and his colleagues (Jackson et al. 1948, 1952, Jackson and Sherman 1953) set up a weathering sequence of clay-size minerals in soils and sedimentary deposits (see Table P.1.2). Pettijohn (1941) compared the frequency of occurrence of each species in recent and older sediments and established an order of persistence, which is in agreement with the Goldich series (see Table P.1.3).

TABLE P.1.1 Mobilities of the Common Cations

→ increasing rate of loss from the environment →	1. Ca^{++} , Mg^{++} , Na^{+} --readily lost under leaching conditions.
	2. K^{+} --readily lost under leaching conditions but rate may be retarded through fixation in the illite structure.
	3. Fe^{++} --rate of loss dependent on the redox potential and degree of leaching.
	4. Si^{4+} --slowly lost under leaching conditions.
	5. Ti^{4+} --may show limited mobility if released from the parent mineral as $\text{Ti}(\text{OH})_4$; if TiO_2 forms, immobile.
	6. Fe^{3+} --immobile under oxidizing conditions.
	7. Al^{3+} --immobile in the pH range of 4.9 to 9.5.

Although still poorly understood, structure must play an important part in the accessibility of waters to the soluble cations. Thus, orthosilicates, e.g., olivine, weather much faster than framework silicates, e.g., feldspars and quartz. However, zircon, also an orthosilicate, is highly resistant to weathering, which indicates that resistance to weathering cannot be based solely on such a simple structural division of the silicates.

The qualitative lists of minerals in Tables P.1.2 and P.1.3 should be quantitatively understood in terms of both thermodynamics (i.e., solubility data) and kinetics (i.e., leaching rates). The solubility and hence the thermodynamic stability of a particular

TABLE P.1.2. Weathering Sequence of Clay-Size Minerals in Soils and Sedimentary Deposits^(a)

Weathering Stage and Symbol	Clay-Size Mineral Occurring at Various Stages of the Weathering Sequence
1, Gp	Gypsum (also halite, etc.)
2, Ct	Calcite (also dolomite, aragonite, etc.)
3, Hr	Olivine-hornblende (also diopside, etc.)
4, Bt	Biotite (also glauconite, chlorite, antigorite, etc.)
5, Ab	Albite (also anorthite, microcline, stilbite, etc.)
6, Qtz	Quartz (also cristobalite, etc.)
7, Il	Illite (also muscovite, sericite, etc.)
8, X	Hydrous Mica - Intermediates
9, Mt	Montmorillonite (also beidellite, etc.)
10, Kl	Kaolinite (also halloysite, etc.)
11, Gb	Gibbsite (also boehmite, etc.)
12, Hm	Hematite (also goethite, limonite, etc.)
13, An	Anatase (also rutile, ilmenite, corundum, etc.)

(a) After Jackson et al. (1948).

TABLE P.1.3. Persistence Order of Minerals^(a,b)

-3. Anatase	11. Epidote
-2. MUSCOVITE	12. HORNBLLENDE
-1. Rutile	13. Andalusite
1. Zircon	14. Topaz
2. Tourmaline	15. Sphene
3. Monazite	16. Zoisite
4. Garnet	17. AUGITE
5. BIOTITE	18. Sillimanite
6. Apatite	19. Hypersthene
7. Ilmenite	20. Diopside
8. Magnetite	21. Actinolite
9. Staurolite	22. OLIVINE
10. Kyanite	

(a) After Pettijohn (1941).

(b) Capitals signify common minerals listed in the Goldrich sequence.

mineral in a weathering solution depend on many environmental factors such as pH, Eh, complexing agents, temperature, fixation/adsorption, ion exchange and ionic strength. These factors are explained briefly below.

- pH. Most minerals are leached faster and have higher solubilities in acid environments. The natural range of possible weathering solutions is pH = 4 to 10. One of the earliest steps in the chemical weathering of a mineral is the exchange of the small and mobile H^+ ion for a cation on the mineral surface, with subsequent disruptions of the structure (Loughnan 1969). Obviously, low pH solutions can accomplish this more effectively.
- Eh. For ions that can exist in several valence states (e.g., U^{+4} and U^{+6}) Eh is very important in determining their solubility. The Eh of natural solutions in contact with the atmosphere is ~ 600 mv. Subsurface solutions can have an Eh range of -400 mv to +400 mv, with the more reducing (low Eh) conditions found in alkaline environments (Garrels and Christ 1965). For example, a mineral with very low solubility, such as uraninite (UO_2), requires a low Eh for stability to weathering (i.e., Eh $< +200$ mv if pH = 6, Eh < 0 mv if pH = 8) (Langmuir 1978).
- Complexing. The formation of complexes has long been recognized as essential in explaining the transport of metals required to form ore deposits. The same must be investigated for the cations of the nuclear waste elements, since complexing can increase the solubility of an element by several orders of magnitude. At lower temperatures ($< 200^\circ C$), we expect carbonates, phosphate, sulfate/sulfide and organic complexes to be important.
- Temperature: The solubility of various minerals can change significantly with temperature. Temperatures in the vicinity of synthetic minerals containing heat producers (^{90}Sr , ^{137}Cs , Actinides) could rise up to several hundred degrees above ambient.
- Adsorption. The ability of ions, such as K^+ , to adsorb strongly to clays and other minerals, retards their mobility and limits their concentration in solution, following leaching of the ions. This may be important, for example, in the case of uranium, which adsorbs strongly to Mn-oxides, Fe-oxides and hydroxides.
- Cation Exchange. An important consideration in establishing the stability of a given nuclear waste element-containing mineral to the leaching of such elements, is the ability of that mineral to exchange the troublesome nuclear waste element for another ion in solution. Thus, K^+ may be exchanged for Cs^+ , or Cl^- may be exchanged for I^- . On the other hand, ion exchange of the radionuclide with clays and other minerals can also retard the mobility of the radionuclide in solution.

Rai and Lindsey (1975) applied simple solubility calculations to deduce the relation between $\log a_{Al}$ and $\log a_{H_4SiO_4}$ at given values of pH, T, and solution compositions (e.g., a_{Ca} , a_{Mg} , etc.) for several aluminosilicates. At a given value of $a_{H_4SiO_4}$ the minerals with the lowest $a_{Al^{3+}}$ were most stable. Using values of $a_{H_4SiO_4}$ typical of soil waters ($a_{H_4SiO_4} \sim 10^{-3.2}$ m) they obtain the stability sequence muscovite > microcline > low albite > anorthite > analcine > pyroxene > K-glass (K-feldspar composition) > Na-glass (albite composition); that is in agreement with Goldich's sequence. Likewise one can plot regions of stability for various minerals on an Eh-pH diagram, as outlined by Garrels and Christ (1965).

P.1.1.2 Use of Geologic Data

Because geologic time spans the lifetimes of the radionuclides of the critical elements, it is very logical to use nature as a laboratory and examine conditions of stability of minerals that may contain the critical elements. In general one recognizes three main geologic environments (igneous, sedimentary, and metamorphic) and asks which mineral phases may exist in each environment and what happens to a mineral grain as it sees a change in its environment. Minerals of the igneous environment see extreme temperatures (and pressures) such that they have crystallized from a melt or a fluid derived from a melt (pegmatites and hydrothermal deposits). The sedimentary environment includes the effect of exposure to the atmosphere and running water and the physical effects of separation and movement of mineral grains. The metamorphic environment involves changing pressure, temperature and pore fluid conditions inducing mineral changes in situ.

As one identifies mineral species that may be potential repository compounds, a test of their stability is to determine the geologic environments under which they can endure. If any modifications in the mineral phase do occur, then the time frame of the modifications can also be deduced. The best test of a mineral's stability is to determine the range of changes through which it can exist.

Many of the minerals that are potentially interesting host phases form initially in the igneous environment. Feldspars, feldspathoids and micas crystallize directly from the melt. Many others are pegmatitic in origin, especially those containing rare earth elements (REE). This information implies conditions that may be necessary to form the phase desired. It may not be the only condition under which the compound will form.

After the compound has formed, the question of what happens to it as the conditions change may be answered. Because stability is the main question, one asks what phase may endure weathering and erosion unchanged, and what new phases are formed if changes do occur. Many minerals survive the rigors of weathering and erosion and these are ultimately collected in detrital deposits. When the detrital deposit has an economic value it is called a placer. These minerals are usually of high density and chemical resistance. Other minerals, called detrital-heavy minerals, may not survive the entire erosion cycle but persist for quite some time. Detrital-heavy minerals may last sufficiently long to allow included radionuclides sufficient time to decay. Therefore, it is useful to identify the placer minerals and other detrital-heavy minerals.

The Placer Minerals

Table P.1.4 identifies the minerals that have been recognized in placer deposits. These minerals are characterized by high densities and chemical and physical resistance. All the noble metals--platinum, iridium, palladium, gold--are known to occur as placer minerals. Many oxides containing lanthanides as well as carbonates, phosphates, tungstates

TABLE P.1.4 Placer Minerals(a)

Element minerals

Platinum, Osmium, Palladium, Iridium, Platiniridium Iridosmine, Osmiridium, Ferroplatinum, Gold, Electrum, Silver, Diamond

Oxide minerals

Tantalite, FeTa_2O_6 ; Thoreaulite, ThTi_2O_6 ; Cassiterite, SnO_2 ; Samarskite, YNb_2O_6 ; Baddeleyite, ZrO_2 ; Euxenite, YNb_2O_6 ; Chromite, FeCr_2O_4 ; Magnetite, Fe_3O_4 ; Columbite, FeNb_2O_6 ; Polycrase, YTi_2O_6 ; Aeschynite, YTi_2O_6 ; Loparite, CeTi_2O_6 ; Ilmenorutile $(\text{Ti,Nb})_3\text{O}_6$; Ilmenite, FeTiO_3 ; Zirkelite, $\text{CaZrTi}_2\text{O}_7$; Pyrochlore, $\text{Ca}_2\text{Nb}_2\text{O}_6\text{OH}$; Rutile, TiO_2 ; Brookite, TiO_2 ; Anatase, TiO_2 ; Corundum, Al_2O_3 ; Spinel, MgAl_2O_4 ; Quartz, SiO_2

Tungstate minerals

Ferberite, FeWO_4 ; Wolframite, $(\text{Fe,Mn})\text{WO}_4$; Hubnerite, MnWO_4 ; Scheelite, CaWO_4

Phosphates

Monazite, CePO_4 ; Xenotime, YPO_4

Carbonates

Bastnaesite, CeCO_3F ; Parisite, $\text{Ce}_2\text{Ca}(\text{CO}_3)_3\text{F}_2$

Silicates

Thorite, ThSiO_4 ; Zircon, ZrSiO_4 ; Garnet, $(\text{Fe,Mg})_3\text{Al}_2\text{Si}_3\text{O}_{12}$; Topaz, $\text{Al}_2\text{SiO}_4\text{F}_2$; Phenakite, Be_2SiO_4

a. Simplified formulae are given. Actual minerals usually contain many additional solid solution substitutions.

and silicates are known placer minerals and therefore are potential lanthanide and actinide phases. Some low density minerals, such as quartz, spinel, garnet, corundum and diamond also occur in placers.

Other minerals might be on this list of placer minerals under special conditions. If the sedimentary conditions are more reducing than usually occurs in nature, uraninite and many sulfide minerals may survive. This possibility is evidenced by the placers of Witwatersrand District of Africa, which formed in the reducing environments of the Pre-Cambrian.

Detrital Minerals

A great many minerals survive long distances of transport in stream beds, although the final fraction of that mineral is often much lower than in the source area. These minerals are listed in Table P.1.5. The rate of degradation of some of these minerals may be sufficiently slow to allow that phase to be a host for radionuclides. Minerals such as apatite, barite, allanite and titanite are particularly interesting. Apatite and allanite contain significant amounts of lanthanides and actinides. Strontium varieties of apatite also occur.

TABLE P.1.5. Detrital Minerals(a)

Elements

Lead

Oxide minerals

Hematite, Fe_2O_3 ; Uraninite, UO_2 ; Uranothorite, $(\text{U,Th})_2$; Leucosene, Ti oxide-hydroxide

Sulfide minerals

Cinnabar, HgS ; Pyrite, FeS_2 ; Marcasite, FeS_2 ; Chalcopyrite, CuFeS_2 ; Arsenopyrite, FeAsS_2 ; Pyrrhotite, Fe_{1-x}S ; Molybdenite, MoS_2 ; Cobaltite, CoAsS_2 ; Dyscrasite, Ag_3Sb ;

Michenerite

PdBiTe ; Geversite, PtSb_2 ; Glaucodot, CoAsS ; Moncheite, PtTe

Sulfate minerals

Barite, BaSO_4

Phosphate minerals

Apatite, $\text{Ca}_5(\text{PO}_4)_3\text{F}$

Silicate minerals

Actinolite, $\text{Ca}_2(\text{Fe,Mg})_5\text{Si}_8\text{O}_{22}(\text{OH})_2$; Andalusite, Al_2SiO_5 ; Biotite, $\text{K}(\text{Fe,Mg})_3\text{AlSi}_3\text{O}_{10}(\text{OH})_2$; Chlorite, $(\text{Mg,Fe})_6(\text{Al,Si})_4\text{O}_{10}(\text{OH})_8$; Chloritoid, $(\text{Fe,Mg})\text{Al}_4\text{Si}_2\text{O}_{10}$; Hornblende, $\text{Ca}_2(\text{Fe,Mg,Al})_5\text{Al}_2\text{Si}_6\text{O}_{22}(\text{OH})_2$; Hypersthene, $(\text{Mg,Fe})\text{SiO}_3$; Kyanite, Al_2SiO_5 ; Olivine, $(\text{Mg,Fe})_2\text{SiO}_4$; Allanite, $\text{Ce}_2\text{Al}_2\text{FeSi}_3\text{O}_{11}(\text{OH})$; Sillimanite, Al_2SiO_5 ; Staurolite, $\text{Fe}_2\text{Al}_9\text{Si}_4\text{O}_{23}\text{OH}$; Titanite, CaTiSiO_5 ; Tourmaline, $\text{Na}(\text{Mg,Fe})_3\text{Al}_6(\text{BO}_3)_3\text{Si}_6\text{O}_{18}(\text{OH})_4$; Zoisite, $\text{Ca}_3\text{Al}_2\text{Si}_3\text{O}_{11}\text{O}(\text{OH})$; Gadolinite, $\text{Be}_2\text{Y}_2\text{FeSi}_2\text{FeSi}_2\text{O}_{10}$

(a) Simplified formulae are given. Actual minerals usually contain many additional solid solution substitutions.

Mineral Associations

In addition to defining regions of stability for specific mineral phases, geologic evidence indicates which phases may occur together in an equilibrium assemblage. These mineral associations are good indicators of compatible phases. The pegmatite environment contains many of the minerals of interest. Rare earth phosphates, rare earth oxides and rare earth carbonates which are good hosts for the lanthanides and actinides, coexist with a variety of complex silicates, which may host other critical elements. These in turn coexist with some of the common silicates, which may be more appropriate hosts for ^{90}Sr and ^{137}Cs .

P.1.2 Kinetic Factors

Often the concentration of an element in solution is not determined by thermodynamic solubility data but by the kinetics of water-rock interactions. Data on this part of the stability criteria are most urgently needed. We outline here the principal factors that indicate the kinetic stability of various minerals.

P.1.2.1 Leaching rate

If the leaching is surface-controlled, the rate at which a cation is leached from a mineral depends on: 1) the reactive specific surface area of the mineral and the solution; 2) the concentrations of the species or ions involved in the transition state of the rate-determining step for surface reaction; 3) the free energy of activation of the activated complex; and 4) the temperature of the solution-rock system. The effects of pH, Eh and complexes enter via their effect on the numbers in 2). The role of temperature in kinetic processes is much more prominent than its role in solubility calculations, due to the high activation energies (10 to 100 Kcal/mole) often encountered. Thus it is crucial to measure accurately the activation energies for the important leaching rates.

Leaching rates can also be controlled by the rate of transport (i.e., diffusion) of leached cations from the weathering mineral-solution interface to the bulk of the solution. In this case, temperature will play a much more minor role, since diffusion activation energies are ~4 to 5 Kcal/mole in electrolyte solutions. Experiments should decide which mechanism is operative for each mineral (e.g., feldspars and calcite seem to weather according to the surface-controlled mechanism, while olivine dissolves by a diffusion-controlled mechanism). The leaching rate may sometimes be severely limited by inhibitors. These inhibitors could be organic substances or ions such as $[\text{PO}_4]^{3-}$, which deactivate the active sites on a surface (e.g., such as the effect of $[\text{PO}_4]^{3-}$ on calcite dissolution). A protective coating may sometimes also form on the surface of the weathering mineral. All these factors add to the kinetic stability of a mineral.

Neither data on leaching rates of relevant minerals nor an understanding on their mechanisms are now available. This gap certainly needs to be filled. The theoretical framework to understand the kinetics of leaching or dissolution is developed to a reasonable degree (Nielsen 1964, Hofmann et al. 1974); however, application to relevant geologic materials is needed.

P.1.3 Crystal Chemical Criteria

P.1.3.1 Element Substitution

In establishing which minerals are appropriate to contain the relevant nuclear waste elements, one may use minerals that are known to contain the element or elements of interest and satisfy the stability criteria. Many such examples will be identified, particularly for Sr, lanthanides, and U. However, elements such as Cs, I, actinides, and Tc are so rare in nature that few known minerals contain them as essential elements. One can then use the principles of crystal chemistry to predict the formation of mineral-like phases that will contain the elements in question or mineral phases into which significant quantities may be incorporated in solid solution.

The critical elements all behave essentially like ions in their compounds, so one can use the principles of element substitution in ionic compounds as criteria for predicting

appropriate host phases. The main criteria are similarity of chemical parameters, particularly the ionic radius and the charge. Other parameters such as polarizability and d-orbital interactions will have a lesser effect in determining the amount of substitution. Thus one can use a table of ionic radii to predict possible substitutions, remembering that charge balance must be maintained by a coupled substitution of another element whenever necessary.

P.1.3.2 Ionic Radii

Table P.1.6 lists the ionic radii of the important nuclear waste elements and of the elements present in minerals which are most likely to be substituted. Usually, complete substitution may occur if the ionic radii differ by no more than 15%. Limited substitution may occur if the radii difference is larger, or a new compound may be induced to form. This compound may be isostructural with the host phase or may have a distinctly different structure. If the phase is isostructural, then stability properties of the new phase may be similar to that of the host, or certainly be close enough to warrant further investigation.

TABLE P.1.6 Selected Ionic Radii (a)

<u>Ion</u>	<u>CN(b)</u>	<u>Ionic Radius (A)</u>	<u>Ion</u>	<u>CN</u>	<u>Ionic Radius (A)</u>
Cs ⁺	X	1.81	Na ¹⁺	VI	1.02
Sr ²⁺	VIII	1.25		IX	1.32
I ¹⁻	VI	2.20	K ¹⁺	VI	1.38
I ⁵⁺	VI	0.95		IX	1.55
Tc ⁴⁺	VI	0.65	Ca ²⁺	VI	1.00
Tc ⁷⁺	VI	0.56		VIII	1.12
La ³⁺	VIII	1.16	Ba ²⁺	VI	1.36
				VIII	1.42
Dy ³⁺	VIII	1.03	Cl ¹⁻	VI	1.81
Ce ⁴⁺	VIII	0.97	Br ¹⁻	VI	1.96
U ⁴⁺	VIII	1.00	Y ³⁺	VIII	1.02
U ⁶⁺	II	0.45	Zr ⁴⁺	VIII	0.84
Np ⁴⁺	VIII	0.98	Ti ⁴⁺	VI	0.61
Pu ³⁺	VI	1.00	Th ⁴⁺	VIII	1.04
Pu ⁴⁺	VIII	0.96	Mn ^{3+(HS)(c)}	VI	0.65
Am ³⁺	VI	1.00	Fe ^{3+(HS)}	VI	0.65
Am ⁴⁺	VIII	0.95	Cr ³⁺	VI	0.62
Cm ³⁺	VI	0.98	Ce ³⁺	VIII	1.11

(a) After Shannon and Prewitt (1969).

(b) CN = coordination number.

(c) HS = high spin.

Using Table P.1.6 as a guide, one can see that Cs^{+1} is large and most like K^{+1} and possibly Ba^{+2} . There is only one mineral in which Cs is essential, and that is pollucite ($\text{Cs}_2\text{Al}_2\text{Si}_4\text{O}_{12} \cdot n\text{H}_2\text{O}$), a member of the analcime ($\text{Na}_2\text{Al}_2\text{Si}_4\text{O}_{12} \cdot n\text{H}_2\text{O}$) family of minerals. The fact that it is acting in the role of Na^{+1} suggests that other Na^{+1} and K^{+1} phases may act as hosts for Cs^{+1} . Other possible examples include the feldspars $(\text{K}, \text{Na}, \text{Ca}) (\text{Al}, \text{Si})_4\text{O}_8$, feldspathoids, $(\text{K}, \text{Na}, \text{Ca}) (\text{Al}, \text{Si})_2\text{O}_3\text{O}_{4-6}$, zeolite, $(\text{K}, \text{Na}, \text{Ca}) \text{Al}, \text{Si})_m\text{O}_{2m} \cdot n\text{H}_2\text{O}$ and micas, $(\text{K}, \text{Na}, \text{Ca})_2(\text{Al}, \text{Mg}, \text{Fe})_{4-6}(\text{Al}, \text{Si})_8\text{O}_{20}(\text{OH})_4$. Traces of cesium are known to occur in each of these minerals.

The next element, Sr^{+2} , is found in many compounds in nature. Often it shows substitutional relations with Ba^{+2} and sometimes with Ca^{+2} . It may also occur in many of the same phases as indicated for Cs^{+1} above.

Iodine exists in nature both as I^- in two compounds and as IO_3^- in several other phases. Its crystal chemistry is similar to the halogens; it behaves most similarly to Br^- and possibly Cl^- , although the radii are markedly different. Very few synthetic iodine compounds have bromine or chlorine isostructural counterparts. Ways to tie iodine up in the crystalline state are discussed later.

Technetium is chemically most similar to manganese and rhenium. There are no known technetium compounds in nature, and there is little knowledge of its crystal chemistry. It is discussed separately below.

The rare earth elements are all very similar in ionic size, although the heavier ones are small enough to cause them to form different series of compounds in some instances from the larger ones. For example, the large lanthanides behave similarly to Ce^{+3} and commonly substitute for it. The smaller lanthanides tend to substitute for Y^{+3} . Rare earths are also known to substitute for Th^{+4} and Zr^{+4} in many of their minerals.

The actinides show some similarities in size and commonly follow Y^{+3} , Th^{+4} , Zr^{+4} , U^{+4} and Ce^{+4} . There are enough differences between uranium chemistry and actinide chemistry to make casual geochemical reasoning suspect and specific research is needed. Uranium readily oxidizes in nature and is commonly found as U^{+6} uranates and as uranyl, UO^{+2} . Plutonyl and Neptonyl can be made and may substitute for uranyl.

Crystalline Solutions

Because of the ease of substitution of ions for other similar ions, it is common for solid solutions to occur. A solid solution is a compound in the crystalline state in which one or more ions have replaced other similar ions in the crystal structure without disrupting the atomic arrangement. Substitutions may be complete (e.g., Fe-Mg in olivine ($\text{Mg}, \text{Fe})_2\text{SiO}_4$), or limited, (e.g., K-Na in nepheline ($\text{Na}, \text{K}) \text{AlSiO}_4$) between two end member compositions.

Natural compounds are rarely pure end members, as solid solution is very common in minerals. Some minerals may have several substitutions and thus extreme variability in chemical compositions occurs. The amphibole family, which has four different sites that allow substitution, is an extreme example. Partial solid solution may actually be desirable

as a waste element fixation mechanism, because the mineral's stability may be better controlled by the host composition. In other words, the waste ion would be sufficiently dilute in the host structure that it does not substantially modify the stability of that host.

Isostructural Compounds

Crystals that allow solid solution necessarily have the same crystal structure for the end members. Compounds with the same structure may show no or very limited solid solubility, usually because of marked size differences of the ions involved. Such isostructural groups may have similar stability properties. Thus it may be useful to identify families of compounds with certain structural properties that may predict the existence of a stable compound of a particular waste element. Calcium compounds, for example, may indicate possible strontium compounds. Bromides and chloride compounds may indicate possible iodide compounds. Several isostructural possibilities are identified below.

P.1.4 Synthesis

Preparation of synthetic minerals requires that the desired elements from the waste streams be mixed with other materials. The mixture is then reacted to form the synthetic mineral. Considered here are the problems that may arise in the processing of nuclear wastes into synthetic minerals.

The purity of the partitioned waste stream will determine whether side reactions will lead to additional phases in the synthetic mineral assemblage. The controlling factors will be the ionic size and the ionic charge of the additional cations present. Ions whose size and charge are similar to those of the element being packaged will dissolve into the synthetic mineral as a minor solid solution. Many of the mineral phases are very "forgiving"; that is, they will accept many elements into solid solution at least in small amounts. If there is a large size or charge mismatch, the impurity elements in the waste stream will react to form secondary minerals of their own. Whether this is detrimental to the processing will have to be evaluated in individual cases.

Three general methods of reaction are in common use among geochemists for the synthesis of minerals: calcination, solid state reaction, and hydrothermal reaction. In each method, it is necessary to mix the waste elements with the other components in the right proportions to form the minerals. Many minerals are nearly stoichiometric, that is the components must be mixed in exactly the proportions called for in the mineral formula. If this is not done, some components will be left over to form additional phases. The stoichiometry of minerals that form solid solutions is not quite so critical.

Mineral synthesis by calcination involves these steps:

- taking each component into solution (for example, as the nitrates)
- mixing the solutions in correct proportions using volumetric methods
- precipitating the solution as a gel, spray drying, or by another method forming a calcine (a highly reactive fine-grained, often poorly-crystallized powder)

- firing the calcine, at temperatures of typically 900° to 1400°C (temperatures depend on the mineral being synthesized) to form the final well-crystallized mineral phase.

The first step is not discussed here since the partitioned wastes are in nitrate solution. Calcination can be carried out using the types of spray calciners that have already undergone considerable development and testing for the solidification of radioactive wastes. No new technology is involved to adapt these devices to synthetic minerals and the expected difficulties are those of remote handling and metering of the solutions and of calciner operation. Firing the calcine to form the final crystalline product in general will require temperatures that can be reached in base metal furnaces or gas-fired kilns.

Mineral synthesis by direct solid-state reaction is done as the name implies. The radioactive waste and the other components needed to construct the mineral phase are mixed as solids. The solid must be intimately mixed, ground, and compacted before reaction. Reaction temperatures are higher and reaction times are longer because the components are crystalline solids and transport can only take place by diffusion. The main difficulty expected here is the maintenance of equipment at the high firing temperatures. There may be more problems with furnace burn-out and breakage or fluxing of refractories. Rare earth and actinide oxides, for example, tend to be very refractory and will require high reaction temperatures if this method is employed.

Hydrothermal synthesis is the technique of reacting materials using high pressure, high temperature water as both a solvent and as a catalyst. It has the tremendous advantage of causing reaction between poorly reactive substances at modest temperatures (200° to 800°C is the experimental range) but it has the important difficulty of requiring reaction at high pressure (hundreds of thousands of atmospheres). To this must be added the difficulties associated with assembling and disassembling the pressure vessel by remote handling. Hydrothermal synthesis is not suited to large scale processing. About the only commercial process that uses hydrothermal synthesis on a large scale is the growth of quartz crystals for the electronics industry. This is a batch process and inherent limitations of pressure vessels require that the batches be fairly small. Commercial quartz-growth vessels are 2 to 3 m high and 0.3 to 0.5 m in diameter.

P.2 DISCUSSION OF MINERAL GROUPS

P.2.1 Silicate Minerals

Silica, SiO_2 , makes up over 60% of the earth's crust, and alumina, Al_2O_3 , makes up another 15%. It is not surprising that these elements dominate the rock-forming minerals. About half of the known mineral species are aluminosilicates, most of which are composed of one or more of the other eleven most abundant elements in the earth's crust. Feldspar alone makes up 58% of the earth's crust. Because of the abundance of these silicate minerals and their occurrence in a wide variety of rocks, one naturally asks if any of them might be potential radionuclide hosts. Detailed chemical and crystallographic data on most of the silicate minerals have been compiled by Deer, Howie, and Zussman (1962).

The suitability of silicates as hosts depends specifically on the ability of the radionuclide to substitute in solid solution for one of the essential ions of the compound. This is especially true for the common rock-forming silicates. We examine each of the major groups of silicate minerals and consider the general principles of crystal chemistry that might elucidate any ionic substitutions of interest. We also consider some common families of silicate minerals that may have potential as repository minerals.

We can dismiss some groups quite easily. The silica (SiO_2) family of minerals is usually rigidly stoichiometric, although substitutions of Al for Si create a charge imbalance; this is usually compensated for by "stuffing" the framework with Na^+ , K^+ or Ca^+ . Cs^+ and Sr^{+2} are too large to enter into these compounds. The olivine-related minerals, including the humite series, are structurally based on close packaging of oxygen ions, and the largest ion that finds its way into these compounds is $(\text{Ca}^{+2})\text{VI}$ at 1.00 Å. Only Tc^{+4} is small enough to fit comfortably, but it is too highly charged. The lanthanide and actinide elements likewise are too highly charged.

P.2.1.1 Pyroxene Minerals

The pyroxene group of minerals are a series of compounds with a general formula $\text{XY}(\text{Si,Al})_2\text{O}_6$, where X represents usually a mono- or di-valent ion with ionic radius in the range 0.6 to 1.0 Å. Examples are Na^+ , Ca^{+2} , Mn^{+2} , Fe^{+2} , Mg^{+2} and Li^+ . The Y cations are di- or tri-valent ions with radii in the range of 0.5 to 0.8 Å. Examples include Mn^{+2} , Fe^{+2} , Mg^{+2} , Fe^{+3} , Al^{+3} , Cr^{+3} , and Ti^{+4} . These small ranges in ionic size result from a structure that is quite closely packed in terms of the oxygen ions. Too much distortion from substitution of larger ions usually breaks down the structure.

About the only critical element which might substitute in pyroxene would be Tc^{+4} with an ionic radius of 0.6 Å. The only other 4-valent ion that occurs in pyroxenes is Ti^{+4} (radius--0.605 Å). Titanium rarely substitutes in quantities greater than one percent by weight, although in some of the titanogites it may reach 3 to 5%.

The suitability of pyroxene as a technetium host requires considerable research and, as a host, pyroxenes are marginal. It is probable that ferrite-like phases will prove more suitable hosts for technetium than any silicate.

The reported rare earth content of any pyroxene is never greater than trace quantities, and these are probably due to minute inclusions of other rare earth minerals.

Pyroxenes form easily in both dry and hydrothermal systems, and they are common reaction products in many silicate experiments. In studies on the decomposition of nuclear waste products in glass under mild hydrothermal conditions, pyroxene was a common end product. Even with the presence of all radionuclides at moderate concentration levels, none of them was detected in the pyroxene phase.

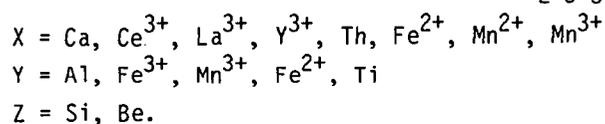
P.2.1.2 Amphibole Minerals

The general formula of the amphibole minerals is $W_{0-1}X_2Y_5(Si_2Al)_8O_{22}(OH)_2$. The X and Y sites are essentially identical with those so labeled in the pyroxene minerals. The limits on ionic substitutions are the same also. The W site, which is not always occupied in amphiboles, accepts low-charge cations in the ionic radius range 0.95_x to 1.35_x Å. These are usually only Na^+ and K^+ , and no other ions are known as substitutes. Amphiboles have sometimes been called "nature's waste-baskets" because the W, X and Y sites can accept so many elements, but the structures are not suitable for any of the critical radionuclides except possibly Tc^{+4} . The remarks concerning Tc^{+4} are the same as for the pyroxenes discussed above.

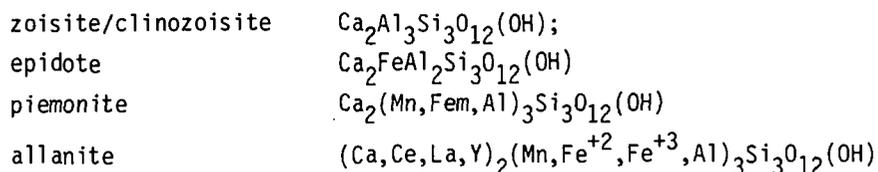
The synthesis of amphiboles is not favorable for them to be considered as potential repository phases. Because the minerals are hydrous, water pressures must be maintained during the synthesis. This, in turn, requires that hydrothermal methods be used. Volcanic rocks rarely contain amphiboles because the water leaves the lava when it reaches the surface. Amphiboles that survive are usually formed in the magma chamber before eruption.

P.2.1.3 Epidote Minerals

The compositional formula for the epidote minerals is $X_2Y_3Z_3(O,OH,F)_{13}$ in which

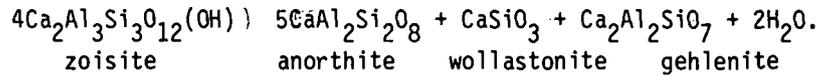


The compositions of epidote minerals that occur commonly are:



Allanite is resistant to weathering and appears as a detrital mineral.

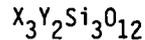
The large X-cation site in epidote is suitable for incorporating ^{90}Sr , rare earths, and possibly actinides in synthetic analogs of allanite. However, epidote is not suitable as a nuclear waste host because of the difficulty in synthesizing the mineral. All of the epidote minerals are stable at low temperatures and modest to high pressure. At high temperature (greater than 600 to 700°C) the epidotes dissociate according to the reaction



Epidote appears readily on a laboratory time scale only at pressures in excess of 3 kilobars and temperatures in the range of 600°C (Deer, Howie, and Zussman 1962). Successful synthesis at atmospheric pressure by calcination or related techniques does not appear likely.

P.2.1.4 Garnet Minerals

The garnets are orthosilicates with the general formula



where X = Mg, Fe²⁺ or Ca; Y = Al, Fe³⁺ or Cr³⁺.

Although the garnets are dense and close-packed structures, the 8-coordinated X-cation site will accept large ions; Sr-substituted grossular (Ca₃Al₂Si₃O₁₂) may fit there. However, grossular is best synthesized at temperatures in the range of 800°C under hydrothermal conditions with a water pressure of 2 kilobars. Attempts at lower pressure synthesis lead to a hydro-garnet, in which OH is substituted for the oxygen, or to mixtures of calcium silicates. In general, garnets are high-pressure phases in nature where they occur in metamorphic rocks. Once formed, the garnets are resistant to weathering and appear as detrital minerals.

P.2.1.5 Calcium Silicate Minerals

Possible candidates among the calcium silicate minerals are limited, partly because of the hydraulic nature of the anhydrous di- and tri-calcium silicates and partly because of the poor resistance of the hydrated phases to mechanical degradation and their high reactivity under quite mild hydrothermal conditions. As with the pyroxenes to which they are related, the structures of possibly useful calcium silicate phases tend to be close-packed with limited possibilities for isomorphous replacement or crystalline solution (at least in the pure phases). Wollastonite (CaSiO₃) and rankinite (Ca₃Si₃O₇) appear the only serious contenders in the group. Both form from oxides at 1200 C and represent the end members of dehydration for hydrated calcium silicate phases. They show little reactivity at lower temperatures; in particular, neither is hydraulic. Strontium can replace calcium in both, making them possible hosts for that cation.

Possibly of more potential use are compounds closely related to the calcium silicates but with off-stoichiometric composition. Bustamite [(Ca,Mn,Fe)SiO₃] and rhodonite [(Mn,Cu)SiO₃], formally allied to wollastonite, have more "open" structures than wollastonite and may be able to accommodate a larger range of foreign ions in substitution. Synthesis and stability of these phases are similar to wollastonite.

Although the pure di-calcium silicates must be ruled out, appreciable amounts of lanthanide solution occurs and stabilizes the non-hydraulic, $\gamma\text{-Ca}_2\text{SiO}_4$ form. This phase may act as strontium and a lanthanide host, but studies are needed to define solubility limits and the stability of material.

Recently, Scott (1976) described the crystal structure of a hydrated potassium-calcium silicate, miserite $[\text{KCa}_5(\text{Si}_8\text{O}_{22})(\text{OH})\text{F}]$, which appears capable of incorporating a wide variety of cations into a vacant site and "locking" them there. The mineral occurs with aegirine and orthoclase, sometimes with wollastonite; it appears to be geologically stable and a potentially useful host for a wide range of cations if some way to incorporate them into structure can be found. Studies of the synthesis and stability of miserite could prove fruitful.

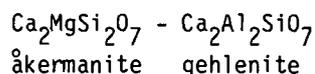
P.2.1.6 Layer Silicate Minerals

The layer silicate minerals include the micas, the clays and the chlorite families. The mica family has the general formula $\text{W}_{0-1}\text{Y}_{2-3}(\text{Si}_2\text{Al})_4\text{O}_{10}(\text{OH})_2$ where W and Y have the same meaning as in the pyroxene and amphibole discussion. The same range of ionic substitutions occurs as in the amphiboles and pyroxenes. Fluorine and less commonly Cl^- and S^{2-} may substitute for the (OH). Biotite is commonly reported from granites and pegmatites, which contain traces of rare earth elements, but these traces can usually be attributed to xenotime $(\text{Y}\dots)\text{PO}_4$ inclusions rather than to being incorporated into the mica structure directly.

The remarks also pertain to the other groups of layer silicates as far as ionic substitutions are concerned. Because chlorites and clays may have layer units with residual electronic charges, some ions may be adsorbed on the surfaces. Interlayer ions may be easily exchanged. The permanence of these attachments, however, is poor and the materials cannot be considered potential repository phases.

P.2.1.7 The Melilite Minerals

The common melilites are a solid solution



in which magnesium is gradually replaced by aluminum. The entire series can be prepared synthetically by dry-firing--that is, calcination techniques at temperatures in the range of 1000 to 1200°C. The minerals as found in nature in high temperature, low pressure environments and synthetically in slags are related materials. They appear to be stable under ambient conditions. Strontium analogs can be made and this mineral series is a potential host for ^{90}Sr .

P.2.1.8 Feldspar Minerals

The feldspar minerals are the most abundant mineral group on the earth and a major constituent of granite rocks, but they are remarkably simple in chemistry. They have the formula $(\text{K,Na,Ca,Ba})_1\text{Al}_{1-2}\text{Si}_{2-3}\text{O}_8$ with almost no other chemical substitutions allowed. Boron

and Fe^{+3} are known to substitute for Al, and Cs and Sr may substitute for the cation. A $\text{SrAl}_2\text{Si}_2\text{O}_8$ phase can be synthesized, which is analogous to $\text{BaAl}_2\text{Si}_2\text{O}_8$, but the level of Sr in natural feldspars is rarely 0.5 wt%. The level of Cs is never greater than 0.005 wt%. Feldspars weather slowly to clay minerals under surface ambients but are very stable in rocks.

P.2.1.9 Feldspathoid Minerals

The feldspathoid minerals form when alkali-rich aluminosilicate compositions have insufficient SiO_2 to form free quartz. The minerals usually coexist with feldspar, particularly the one with the corresponding alkali ion. The important feldspathoids are nepheline, $(\text{Na},\text{K})_4\text{Al}_4\text{Si}_4\text{O}_{16}$; leucite, KAlSi_2O_6 ; analcime, $\text{NaAlSi}_2\text{O}_6 \cdot \text{H}_2\text{O}$; sodalite, $\text{Na}_8\text{Al}_6\text{Si}_6\text{O}_{24}\text{Cl}_2$, and cancrinite $(\text{Na},\text{K},\text{Ca})_{6-8}(\text{Al},\text{Si})_{12}\text{O}_{24}(\text{Cl},\text{SO}_4,\text{CO}_3)_{1.5-2.0} \cdot \text{nH}_2\text{O}$. Scapolite, $(\text{Na},\text{Ca},\text{K})_4\text{Al}_3\text{Ial},\text{Si})_3\text{Si}_6\text{O}_{24}(\text{Cl},\text{SO}_4,\text{CO}_3)$, may also be considered here because it resembles sodalite and cancrinite in behavior although it is not formally considered a feldspathoid.

Nepheline is a stuffed derivative of tridymite (SiO_2) and can accept alkali ions in the framework to charge compensate the Al that substitutes for Si. The cages are just large enough to accept K (ionic radius = 1.38 Å) and actually prefer some Na (ionic radius = 1.02 Å) to relieve some of the strains on the framework linkages. To accept larger cations such as Cs and Sr would be too much strain on the structure. Cs and Sr are generally not reported in any nepheline analyses.

Leucite and analcime have similar crystal structures with identical frameworks. The cages are larger than in nepheline and Cs will substitute freely in the analcime to form the only Cs mineral in nature. Pollucite, $\text{CsAlSi}_2\text{O}_6 \cdot 0.5\text{H}_2\text{O}$, forms readily from its components, and is the leading candidate as a repository phase for Cs (Komarnini et al. 1978). Considerable study has already been made on pollucite for this purpose. The possibility of a Sr analog also exists, but it does not occur in nature.

Sodalite, cancrinite and scapolite may have two uses as potential waste minerals although considerable research is needed to verify their potential. All three minerals may have Cs and Sr analogs, where these elements substitute for Na, Ca, or K, as in leucite-analcime. The framework cages are larger than in leucite and analcime, but because of this increased size the alkali cations are easily exchanged and hence easily leachable. Another interesting aspect of these structures is the trapping of large anions in the cages. All three minerals are known to have significant quantities of Cl^- , SO_4^{2-} and CO_3^{2-} in the structural cages, and sodalite often has S^{2-} . This behavior immediately suggests the possibility of trapping I^- inside the cages. If the structure can be grown around the I^- before the iodine volatilizes, it may be effectively caged because its radius (2.20 Å) is considerably larger than the cage opening (1.40 Å). Much research is needed on this potential.

P.2.1.10 Zeolite Minerals

The zeolites are a large group of industrially important compounds, many of which exist as minerals. Their properties have been surveyed by Breck (1959). They have

aluminosilicate framework structures with larger cages and cage openings than do the feldspathoids, and all zeolites show exchange properties of the nonframework cations. This property is undesirable in a repository compound unless the radionuclide can be stabilized in the structure.

Both Cs and Sr zeolites have been synthesized, and one Sr zeolite occurs in nature, the mineral brewsterite, $\text{SrAl}_2\text{Si}_6\text{O}_{16} \cdot 5\text{H}_2\text{O}$. It is found in volcanic basalts in gas cavities as a very late-formed mineral.

Zeolites can be synthesized by gel and by hydrothermal methods. They contain considerable water, which helps keep the framework open and which can be driven off by heat. Some structures collapse at relatively low temperatures, even as low as 100°C ; but may retain their structural integrity as high as 800°C . The exchangeability of the cation, however, suggests that the zeolites in general will not desired cations for sufficient times under various conditions to be effective repository compounds.

Rare earths have been exchanged in some of the zeolite phases. In particular the faujasite series may be synthesized with a Ce:Ca ratio of 6:4 (Olsen et al. 1967). The faujasites have one of the more open zeolite framework structures. Considerable research is needed to determine the suitability of zeolite structures as waste repositories; they cannot be dismissed summarily.

P.2.11 Borosilicate Minerals

Because boron forms a very stable oxyanion, both as BO_3 and BO_4 coordination polyhedra, many borosilicates are quite stable mineral structures. Beryllium as BeO_4 coordination polyhedra also forms quite stable minerals with silicates. Many minerals of this type are known to contain rare earth elements (REE) either as essential elements or in solid solution to significant levels. Table P.2.1 lists the most important of these minerals. These minerals are considered possible repository phases.

The borosilicates and berylosilicates are primarily found in rare-earth bearing pegmatites, both granite and nepheline syenite types. The affinity for rare-earth elements is indicated by their formation. The stability of these phases under repository conditions is unknown. Considerable experimentation is needed to determine their suitability.

P.2.1.12 Zirconosilicate and Titanosilicate Minerals

Interest in the zirconosilicate and titanosilicate minerals arises from the known substitution of rare-earth elements and actinides for both Ti and Zr. Usually, the quantities are small. The known minerals are listed in Table P.2.2. Both the zirconosilicates and titanosilicates are formed in pegmatite deposits. They are commonly associated with other rare-earth bearing minerals. Evidence suggests that many of them may be quite resistant to weathering and zircon and titanite are known to survive as heavy minerals in placer deposits.

TABLE P.2.1 Borosilicate and Berylosilicate Minerals

Borosilicates	Formula
Cappelenite	$(\text{Ba,Ca,Na})(\text{Y,La})_6\text{B}_6\text{Si}_{13}(\text{O,OH})_{27}$
Danburite	$\text{CaB}_2\text{Si}_2\text{O}_8$
Hellandite	$(\text{Ca,Y})_2(\text{Si,B,Al})_3\text{O}_8 \cdot \text{H}_2\text{O}$
Melanocerite	$(\text{Ce,Ca})_5(\text{Si,B})_3\text{O}_{12}(\text{OH,F}) \cdot n\text{H}_2\text{O}$
Stillwellite	$(\text{Ce,La,Ca})\text{BSiO}_5$
Tadzhikite	$\text{Ca}_3(\text{Ce,Y})_2(\text{Ti,Al,Fe})\text{B}_4\text{Si}_4\text{O}_{22}$
Tourmaline	$(\text{Na,Ca})(\text{Mg,Fe...})_3\text{Al}_6(\text{BO}_3)_3(\text{Si}_6\text{O}_{18})(\text{OH,F})_4$
Tritomite	$(\text{Ce,La,Y,Th})_5(\text{Si,B})_3(\text{O,OH,F})_{13}$
Tinzenite	$(\text{Ca,Mn,Fe})_3\text{Al}_2\text{BSi}_4\text{O}_{15}(\text{OH})$
Berylosilicates	Formula
Aminoffite	$\text{Ca}_2(\text{Be,Al})\text{Si}_2\text{O}_7(\text{OH}) \cdot \text{H}_2\text{O}$
Gadolinite	$\text{Be}_2\text{Y}_2\text{FeSi}_2\text{O}_{10}$
Semenovite	$(\text{Ca,Ce,La})_{12}(\text{Be,Si})_8\text{Si}_{12}\text{O}_{40}(\text{O,OH,F})_8 \cdot \text{H}_2\text{O}$
Tugtupite	$\text{Na}_4\text{AlBeSi}_4\text{O}_{12}\text{Cl}$

TABLE P.2.2. Zirconosilicate and Titanosilicate Minerals

Zirconosilicates	Formula
Armstrongite	$\text{CaZrSi}_6\text{O}_{15} \cdot 2.5\text{H}_2\text{O}$
Bazirite	$\text{BaZrSi}_3\text{O}_9$
Catapleiite	$\text{Na}_2\text{ZrSi}_3\text{O}_9 \cdot 2\text{H}_2\text{O}$
Elpidite	$\text{Na}_2\text{ZrSi}_6\text{O}_{15} \cdot 3\text{H}_2\text{O}$
Eudialyte	$\text{Na}_4(\text{Ca,Ce,Fe})_2\text{ZrSi}_6\text{O}_{17}(\text{OH,Cl})_2$
Hilairite	$\text{Na}_2\text{ZrSi}_3\text{O}_9 \cdot 3\text{H}_2\text{O}$
Lavenite	$(\text{Na,Ca})_3\text{ZrSi}_2\text{O}_7(\text{O,OH,F})_2$
Lemoyneite	$(\text{Na,Ca})_3\text{Zr}_2\text{Si}_{10}\text{O}_{26} \cdot 8\text{H}_2\text{O}$
Vlasovite	$\text{Na}_2\text{ZrSi}_4\text{O}_{11}$
Wadeite	$\text{K}_2\text{ZrSi}_3\text{O}_9$
Zircon	ZrSiO_4
Titanosilicates	Formula
Batisite	$\text{Na}_2\text{BaTi}_2\text{Si}_4\text{O}_{14}$
Chevkinite	$(\text{Ca,Ce,Th})_4(\text{Fe,Mg})_2(\text{Ti,Fe})_3\text{Si}_4\text{O}_{22}$
Ilmajokite	$(\text{Na,Ba,Ce})_{10}\text{Ti}_5\text{Si}_{14}\text{O}_{22}(\text{OH})_{44} \cdot n\text{H}_2\text{O}$
Joaquinite	$\text{Ba}_2\text{NaCe}_2\text{Fe}(\text{Ti,Nb})_2\text{Si}_8\text{O}_{26}(\text{OH,F})$
Karnasurtite	$(\text{Ce,La,Th})(\text{Ti,Nb})(\text{Al,Fe})(\text{Si,P})_2\text{O}_7(\text{OH})_4 \cdot 3\text{H}_2\text{O}$
Lamprophyllite	$\text{Na}_2(\text{Sr,Ba})_2\text{Ti}_3(\text{SiO}_4)_4(\text{OH,F})_2$
Mosandrite	$(\text{Na,Ca,Ce})_3\text{TiSi}_2\text{O}_8\text{F}$
Perrierite	$(\text{Ca,Ce,Th})_4(\text{Mg,Fe})_2(\text{Ti,Fe})_3\text{Si}_4\text{O}_{22}$
Titanite	CaTiSiO_5
Tranguillityite	$\text{Fe}_8(\text{Zr,Y})_2\text{Ti}_3\text{Si}_3\text{O}_{24}$
Tundrite	$\text{Na}_3(\text{Ce,La})_4(\text{Ti,Nb})_2(\text{SiO}_4)_2(\text{CO}_3)_3\text{O}_4(\text{OH}) \cdot 2\text{H}_2\text{O}$

The stability of this group of minerals under repository conditions deserves more study. They may actually prove to accept Cs^{+1} and Sr^{+2} in some of their structures for Ca, Na, or Ba. One Sr phase, lamprophyllite, is known.

P.2.1.13 Rare-Earth Silicate Minerals

A large number of minerals are essentially rare-earth silicates with or without other essential elements. These compounds must all be considered potential repository phases for both the lanthanides and actinides. Some of the phases have demonstrated stabilities, having formed in granites or pegmatites and then survived the sedimentary cycle to be deposited in Placers. Allanite is one example; it was discussed with the epidote minerals. Thorite, huttonite, and cheralite are other examples.

Most of the minerals in Table P.2.3 are formed in pegmatites. The lanthanide (Ln) families of $\text{Ln}_2\text{Si}_2\text{O}_7$ and Ln_2SiO_5 phases are easy to prepare synthetically. Many of them show several structural modifications, but they have high melting or decomposition temperatures. Some of the minerals such as coffinite, USiO_4 , may be synthesized at 100°C . These minerals form in sedimentary rocks from circulating ground waters.

TABLE P.2.3. Rare-Earth Silicate Minerals

Rare-Earth Silicates	Formula
Allanite	$(\text{Ce, Ca, Y})_2(\text{Fe, Al})_3(\text{SiO}_4)_3(\text{OH})$
Ashcroftite	$\text{KNaCaY}_2\text{Si}_6\text{O}_{12}(\text{OH})_{10} \cdot 4\text{H}_2\text{O}$
Britholite	$(\text{Ca, Ce})_5(\text{SiO}_4, \text{PO}_4)_3(\text{OH, F})$
Cheralite	$(\text{Ca, Ce, Th})(\text{P, Si})\text{O}_4$
Coffinite	$\text{U}(\text{SiO}_4)_{1-x}(\text{OH})_{4x}$
Ekanite	$(\text{Th, U})(\text{Ca, Fe, Pb})_2\text{Si}_8\text{O}_{20}$
Huttonite	ThSiO_4
Iimorite	$\text{Y}_5(\text{SiO}_4)_3(\text{OH})_3$
Kainosite	$\text{Ca}_2(\text{Ce, Y})_2\text{Si}_4\text{O}_{12}(\text{CO}_3) \cdot \text{H}_2\text{O}$
Miserite	$\text{K}(\text{Ca, Ce})_4\text{Si}_5\text{O}_{13}(\text{OH})_3$
Nordite	$(\text{La, Ce})(\text{Sr, Ca})\text{Na}_2(\text{Na, Mn})(\text{Zn, Mg})\text{Si}_6\text{O}_{17}$
Phosinaite	$\text{H}_2\text{Na}_3(\text{Ca, Ce})\text{SiO}_4\text{PO}_4$
Sazhinite	$\text{Na}_3\text{CeSi}_6\text{O}_{15} \cdot 6\text{H}_2\text{O}$
Soddyite	$(\text{UO}_2)_5\text{Si}_2\text{O}_9 \cdot 6\text{H}_2\text{O}$
Thalenite	$\text{Y}_2\text{Si}_2\text{O}_7$
Thorite	ThSiO_4
Thorosteenstrupine	$(\text{Ca, Th, Mn})_3\text{Si}_4\text{O}_{11}\text{F} \cdot 6\text{H}_2\text{O}$
Thörtveitite	$(\text{Sc, Y})_2\text{Si}_2\text{O}_7$
Tombarthite	$\text{Y}_4(\text{Si, H}_4)_4\text{O}_{12-x}(\text{OH})_{4+2x}$
Tornebohmite	$(\text{Ce, La})_3\text{Si}_2\text{O}_8(\text{OH})$
Umbozerite	$\text{Na}_3\text{Sr}_4\text{ThSi}_8(\text{O, OH})_{24}$
Uranophane	$\text{Ca}(\text{UO}_2)_2(\text{SiO}_3\text{OH})_2$
Weeksite	$\text{K}_2(\text{UO}_2)_2\text{Si}_6\text{O}_{15} \cdot 4\text{H}_2\text{O}$
Yttrialite	$(\text{Y, Th})_2\text{Si}_2\text{O}_7$

close association with albite, zircon, apatite, sphene, biotite. Pyrochlore occurs in both the metamict and crystalline state. It has quite a variety of names:

Pyrochlore $(\text{Na,Ca,U,Ce,Y})_{2-x}(\text{Nb,Ta,Ti})_2\text{O}_6(\text{OH,F})$;
 Betafite $(\text{U,Ca})_{2-x}(\text{Nb,Ti,Ta})_2\text{O}_{6-x}(\text{OH})_{1+x}$, high Ti and U;
 Zirconolite $(\text{CaZrTi}_2\text{O}_7)$;
 Microlite $(\text{Ca,Na})_2\text{Ta}_2\text{O}_6(\text{O,OH,F})$, high Ta;
 Djalmaite $(\text{Ca,Na,U})_2\text{Ta}_2\text{O}_6(\text{O,OH,F})$, high U relative to microlite;
 Obruchinite $(\text{Y,U,Ca})_{2-x}\text{Nb}_2\text{O}_6(\text{OH})$, low Ti, high Y and U.

The differences among minerals reflect only the amounts of U, Ti, Ta, Y relative to pyrochlore. Pyrochlore from carbonatites can have up to 4% ThO_2 . Hydration of pyrochlore leads to loss of mobile REE, Ca, Na and an increase in U (Vlasov 1966). Pyrochlore can have up to 19% U_3O_8 and high Sr. Pyrochlore also occurs as a placer deposit-forming mineral.

P.2.2.3 AB_2O_6 --Nb-Ti-Ta Oxides

Columbite Structure

Columbite $(\text{Fe,Mn})(\text{Nb,Ta})_2\text{O}_6$ (tantalite). Columbite can have up to 3% REE, little U.

It is very abundant in acid rocks, e.g. (rarer) granite, granitic pegmatites, quartz veins; occurs in association with biotite, albite, zircon. Columbite-tantalite is a placer deposit-mineral and is insoluble in acids (Vlasov 1966). Furthermore, it is very resistant to weathering and accumulates in deluvial, eluvial, and alluvial placers, resulting from the weathering of columbite-bearing granite and pegmatite. In placers, it is associated with cassiterite, zircon, ilmenite and rutile. Columbite may be a good candidate for hosting lanthanide and actinide elements.

Euxenite Structure

Euxenite-polycrase $\text{Y}(\text{Nb,Ti})_2(\text{O,OH})_6$ -- $\text{Y}(\text{Ti,Nb})_2(\text{O,OH})_6$

Delorenjite $\text{Y}(\text{Ta,Nb})_2(\text{O,OH})_6$

Fersmite $(\text{Ca,Ce})(\text{Nb,Ti,Fe})_2(\text{O,OH,F})_6$

Thorium is in higher coordination in euxenite structures than in columbite structures. Th, U, and Ca can replace Y up to several percent, U up to 16% UO_2 , Th up to 8% ThO_2 . Euxenites are widespread in granite pegmatites. Euxenite occurs as accessory mineral in granites and is also found in small amounts in placers. It is associated with ilmenite, monazite, xenotime, zircon, and garnet.

Fersmite is found in nepheline-syenite and carbonatite massifs in association with columbite, apatite, calcite, fluorite. It is typical of rocks of intermediate composition (for weathering and alteration see below).

Priorite Structure

Priorite-Aeschynite $(\text{Ce,Nd,Th,Y})(\text{Ti,Nb})_2\text{O}_6$

Polymignite $(\text{Ca,Fe,Ce})(\text{Zr,Ti,NbO}_2)_2\text{O}_6$

Sinicite $(\text{Ce,Nd,Th,U})(\text{Ti,NbO}_2)_2\text{O}_6$, high U.

Priorite differs from euxenite by having cerium REEs and a high content of thorium and Zr (little U). The REE have the same coordination as in the euxenite structure. Aeschynite occurs as an accessory in some deposits related to nepheline-syenite and alkali-syenite massifs in association with zircon, biotite, corundum, muscovite, sphene, and fluorite.

The weathering and alteration of the AB_2O_6 and $A_2B_2O_6(O,OH,F)$ REE-Nb-Ti-Ta complex oxides can be handled in one group. These oxides have pervasive alteration with a usual weathered crust surrounding fresher oxides (Ewing 1975a). The results of weathering are leaching of the A-site cations (i.e., U, REE) and introduction of H_2O or OH^- or $O^{=}$ into the oxide. The B cation remain basically unchanged (Ewing 1975a, Wambeke 1970).

In weathering, up to 40% decrease in the REE content is possible, although the REE distributions remain nearly the same (Ewing 1975a). For example, a priorite from the Kibara Mountains, North Katanga, had a fresh inner zone (black) with ~ 0.075 cerium atoms and 0.95 U atoms per 5.58 O atoms. Wambeke (1970) gives the relative leaching rate of A cations as 110 REE atoms, 120 Na atoms and 40 U atoms per 100 atoms of Ca leached out. There are little hard data on the kinetics or solubility of these complex oxides; these should be obtained. It seems that columbite might be a good candidate among this group for Ce disposal, since it can be very resistant to alteration. Euxenite is the candidate for the U elements.

P.2.2.4 ABO₄ Oxides

Fergusonite Structure

A = Y, REE, U, Ca, Th

B = Nb, Ta, Ti.

Solid solution: $YNbO_4$ - $YTaO_4$
fergusonite formanite

The REE in fergusonite are mostly the yttrium rare earths (Vlasov 1966). Fergusonite occurs as a metamict mineral. Fairly abundant in granite pegmatites, it accumulates in small amounts in placers and is found as an accessory mineral in granites. In pegmatites, it is associated with zircon, monazite, xenotime and euxenite. A study of monazite-bearing alluvial deposits in Malaya (Flinter et al. 1963) showed fergusonite occurring with columbite, Ta/Nb rutile, cassiterite and garnet. The samples were derived from a cassiterite-bearing granite. It thus seems that fergusonite might be relatively stable as a host of REE and actinides.

P.2.3 Carbonate and Sulfate Minerals

P.2.3.1 Rare Earth Fluorocarbonates

Carbonate minerals are compounds of some cations with the carbonate anion, CO_3^{2-} , often with hydroxyls and waters of hydration. Of more than 70 naturally occurring carbonate compounds, most are either water soluble or are easily decomposed. These include the simple and

complex carbonates of the alkali metals, the alkaline earth metals, and the transition metals. Most carbonates are sensitive to pH and dissolve easily in low pH solutions.

Exceptions to the general instability of carbonate minerals are the fluorocarbonate compounds of the rare earths. These are:

Bastnaesite	$(\text{Ce,La})\text{CO}_3\text{F}$
Parisite	$\text{Ca}(\text{Ce,La})_2(\text{CO}_3)_3\text{F}_2$
Cordylite	$\text{Ba}(\text{Ce,La})_2(\text{CO}_3)_3\text{F}_2$
Synchisite	$\text{Ca}(\text{Ce,La})(\text{CO}_3)_2\text{F}$

Bastnaesite and parisite are relatively insoluble even in low pH solutions at ambient temperatures. None are insoluble in hot, low pH solutions. The rare-earth fluorocarbonates could act as hosts for rare-earth elements in neutral or alkaline repository rocks.

P.2.3.2 Sulfate Minerals

The number of sulfate minerals numbers several hundred but nearly all are soluble in water or are otherwise unstable. Two exceptions of interest are barite, BaSO_4 , and celestine, SrSO_4 . The solubility of barite in cold water is only 2.2 ppm while the solubility of celestine is 113 ppm. There is a complete solid solution between barite and celestine although intermediate compositions are not found in nature.

Use of barite and celestine as hosts for ^{90}Sr would be of value in a bedded anhydrite repository (anhydrite = CaSO_4) because of the chemical compatibility.

P.2.4 Phosphate Minerals

Natural phosphate minerals are all orthophosphates, the major one being fluorapatite. The phosphate-containing minerals include a subset, that seems particularly suited to the disposal of nuclear waste elements: the apatite family and the monazite-xenotime family.

Since in nature phosphorus will exist in only one valence state (+5) (for example, $\text{H}_2\text{PO}_3^- > \text{H}_2\text{PO}_4^-$ only when $f\text{O}_2 < 10^{-101}$ at 25°C), the distribution and stability of its species in solution will be Eh-independent. On the other hand, the dominant phosphorus species in solution will be strongly dependent on pH and on possible complexing cations, since PO_4^{3-} , HPO_4^{2-} and H_2PO_4^- form strong complexes [e.g. with uranium (Langmuir 1978)]. The reaction



has a $\Delta G_r^\circ = 9.83$ kcal/mole and a $\Delta H_r^\circ = +0.99$ kcal/mole at 25°C , which yields a $K_1 = 10^{-7.21}$ at 25°C . Hence, for $\text{pH} < 7.21$, H_2PO_4^- will be the dominant PO_4 species in solution and for $\text{pH} > 7.21$, HPO_4^{2-} will be dominant. Ignoring complexes, this will also be true at higher temperatures, since ΔH_r° is so small. The total phosphorus content of ground waters, ΣPO_4 , is most often greater than 0.1 ppm but rarely greater than 1 ppm.

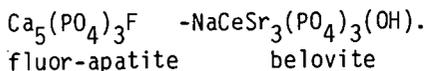
In the mineral structure, the PO_4 tetrahedra can often be replaced by the CO_3 , SO_4 , and SiO_4 groups leading to a variety of phosphate minerals.

P.2.4.1 Apatite Family-Ca₅(PO₄)₃(OH,F)

Apatite is the most abundant phosphorus-bearing mineral. It is a common accessory mineral in many types of rocks (acid to basic). Apatite can take up significant amounts of Sr (up to 11.6 wt% SrO) and also rare earths (up to 11 wt% REE) and so may be a suitable host for nuclear waste elements. The rare earths, predominantly Ce, may replace Ca in apatites of alkaline igneous rocks. U⁴⁺ (r = 0.97 Å) can also substitute for Ca²⁺ (r = 0.99 Å). Natural apatites have ~0.01% U, if primary igneous apatite, or slightly richer; 0.02% U if sedimentary marine apatite. Thorium is more abundant than U by a factor of 3 or 4 (Deer 1962). Apatites can contain CO₃, SO₄ and SiO₄ groups replacing PO₄. In sedimentary phosphorites, the apatite can have up to 7 to 8% CO₃ content, with much lesser SO₄ or SiO₄ substitution. The carbonate content of onshore phosphorites is less (3%) than that of sea floor phosphorites, suggesting that weathering reduces the carbonate content.

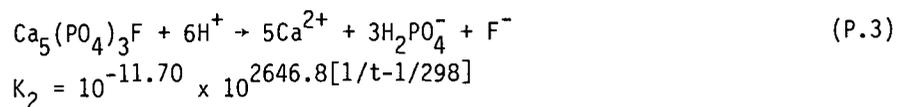
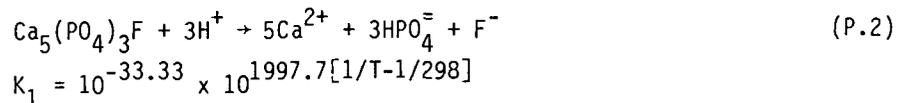
In terms of geologic evidence for stability to weathering apatite is not uncommon in sedimentary rocks where it occurs both as a detrital mineral and as a primary deposit. It is not classified as a placer deposit-forming mineral, however. On the weathering stability list of Pettijohn (1941), apatite has an index of 6, putting it beneath biotite and garnet. Smithson (1941) from a study of Jurassic sandstones in Yorkshire, England, lists apatite as stable in unweathered rock but decomposed in weathered rocks. Graham (1950) lists apatite with olivine as least stable and Jackson (1953) puts it low in the second stage of the weathering sequence of clay-size mineral particles. Thus, the stability of apatite has yet to be firmly shown.

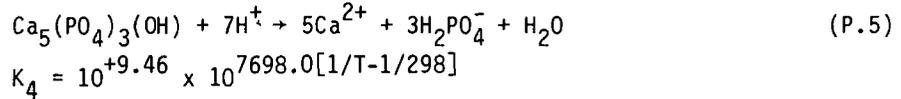
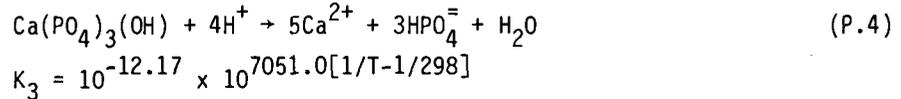
Strontium apatite results in the solid solution:



However, belovite is unstable under surface conditions and is readily replaced by rhabdophanite, CePO₄ · H₂O; Sr and Na are then rapidly lost (Vlasov 1966). There is unlimited substitution in the systems Ca₅(PO₄)₃F-Sr₅(PO₄)₃F and Ca₅(PO₄)₃(OH)-Sr₅(PO₄)₃(OH). Sr-apatite, found in alkali pegmatites, is readily soluble in acids (Vlasov 1966).

We can use the solubility criteria laid out in the introduction to this appendix to examine the stability of apatite minerals. Although thermodynamic data for Sr-apatite are lacking, there are data for fluorand hydroxy-apatite (Naumov et al. 1974). Using these data, we can compute the following:





If we use $\Sigma \text{PO}_4 = 10^{-6}\text{m}$ (~ 0.1 ppm) and $a_{\text{F}^-} = 1.6 \times 10^{-5}\text{m}$ (~ 0.3 ppm), typical values for ground waters, we obtain the following values for the activity of calcium in equilibrium with the apatites:

Fluor-apatite

$a_{\text{Ca}^{2+}}$		
pH/T	25°C	75°C
6	$1.04 \times 10^{-5}\text{m}$	$5.81 \times 10^{-6}\text{m}$
8	$1.24 \times 10^{-7}\text{m}$	$7.95 \times 10^{-8}\text{m}$

Hydroxy-apatite

$a_{\text{Ca}^{2+}}$		
pH/T	25°C	75°C
6	$1.23 \times 10^{-3}\text{m}$	$2.24 \times 10^{-4}\text{m}$
8	$5.83 \times 10^{-6}\text{m}$	$1.22 \times 10^{-6}\text{m}$

In ground waters, $a_{\text{Ca}^{2+}}$ is typically $\sim 10^{-3}\text{m}$ (Rai and Lindsay 1975). Therefore in alkaline environments we expect both apatites to be stable at temperatures from 25°C to 100°C. However, in acid environments hydroxy-apatite will not be stable, while fluor-apatite will be somewhat stable, more so at higher temperatures. Chien (1977) has also shown that the carbonate substitution may increase the equilibrium dissolution of apatite.

P.2.4.2 Monazite-Xenotime Family--(Ce,La)PO₄-YPO₄

This family is one of the most promising for the disposal of nuclear wastes. Both monazite, $(\text{Ce,La})\text{PO}_4$ and xenotime, YPO_4 , as well as their hydrates, rhabdophanite, $(\text{Ce,Ca})\text{PO}_4 \cdot \text{H}_2\text{O}$, and churchite, $\text{YPO}_4 \cdot \text{H}_2\text{O}$, are simple orthophosphates. They are always crystalline even though they may contain significant amounts of U and Th. Monazite is isostructural with huttonite, ThSiO_4 , and xenotime is isostructural with zircon (ZrSiO_4) and coffinite (USiO_4). Monazite can acquire quite a high content of thorium (28%) by the substitution $\text{Th}^{4+} + \text{Si}^{4+} \rightarrow \text{Ce}^{3+} + \text{P}^{5+}$ (i.e., ThSiO_4 - CePO_4 solid solution). Monazite is a selective cerium mineral (i.e., large-radius rare earths). It has lesser amounts of uranium (up to 4%) (Deer et al. 1962, Vlasov 1966). It is sparingly soluble in acids and is very stable under weathering conditions, often collecting in placers formed from the disintegration of monazite-containing granites. It occurs as an accessory in granites and granitic pegmatites and is abundant in metamorphic deposits (Vlasov 1966). It occurs as a detrital mineral in sands from weathering of granites and gneisses.

Dryden and Dryden (1946) compared the changes in relative abundance of various minerals from the fresh rocks to the weathered products in samples from the Wissahickon schist in Pennsylvania and Maryland. They found, by taking the ratios of the number of grains of each mineral in fresh and weathered rock, that the resistance of zircon relative to garnet is 100 (i.e., $\text{garnet}/\text{Zr}_{\text{fresh}}/\text{garnet}/\text{Zr}_{\text{weathered}} \sim 100$), sillimanite 40, monazite 40, chloritoid 20, kyanite 7 and all other minerals less than 5. This is in agreement with Pettijohn (1941) who ranked monazite in his "weathering sequence" as 3 after zircon (1) and tourmaline (2). The general geologic evidence points to a very resistant mineral.

We can calculate the solubilities for monazite to establish its thermodynamic stability. Taking $\Sigma\text{PO}_4 = 10^{-6}\text{m}$ (0.1 ppm), we can compute the solubility of Ce^{3+} in a natural leaching solution as a function of pH and temperature. The thermodynamic data for CePO_4 were obtained from Naumov, et al. (1974). We obtain:



$$\Delta G_r^\circ = 3.27 \text{ kcal/mole} \quad \Delta H_r^\circ = -11.71 \text{ kcal/mole}$$



$$\Delta G_r^\circ = 12.10 \text{ kcal/mole} \quad \Delta H_r^\circ = -10.72 \text{ kcal/mole.}$$

Therefore

$$K_6 = 4.00 \times 10^{-3} e^{5893.3[1/T-1/298]}$$

$$K_7 = 2.46 \times 10^{-10} e^{5395.1[1/T-1/298]}$$

Assuming no complexing, pure solids, and $\Sigma\text{PO}_4 = 10^{-6}\text{m}$, then

$$a_{\text{Ce}^{3+}}(\text{aq})$$

pH/T	25°C	50°C
6	$4.0 \times 10^{-9}\text{m}$	$8.6 \times 10^{-10}\text{m}$
8	$2.46 \times 10^{-12}\text{m}$	$6.1 \times 10^{-13}\text{m}$

The low values of $a_{\text{Ce}^{3+}}$ obtained support the stability evidence from the geologic data. Obviously monazite is more stable in warm alkaline environments. Increasing the phosphate content of the ground water would also further stabilize the monazite. Thus if $\Sigma\text{PO}_4 = 10^{-5}\text{m}$ (1 ppm), $a_{\text{Ce}^{3+}} = 4.0 \times 10^{-10}\text{m}$ at pH = 6, T = 25°C and the same for the other conditions.

Xenotime contains a high amount of yttrium rare earths. It is widespread in granites, pegmaties and metamorphic gneisses (Vlasov 1966). When granites weather, xenotime accumulates in placers (e.g. in New Zealand and USSR). Xenotime is very stable under surface conditions.

P.2.5 Iodine Hosts

P.2.5.1 Iodine Minerals

Iodine is a relatively rare element in rocks and minerals. It occurs in both the I^- and I^{5+} valence states. Iodine is easily oxidized to the 5-valent state and appears in many of its natural compounds as the iodate, IO_3^- ion. These are:

Lautarite	$Ca(IO_3)_2$
Bellingerite	$Ca(IO_3)_2 \cdot 2/3H_2O$
Salesite	$Cu(IO_3)OH$
Schwartzembergite	$Pb_5(IO_3)Cl_3O_3$
Dietzeite	$Ca_2(IO_3)_2CrO_4$

The above compounds are at least slightly soluble in water, and all are soluble in solutions with low pH. The iodate minerals are found in evaporite deposits or as weathering products of ores in very dry environments.

Marshite, CuI , iodargyrite, AgI , and their solid solution, miersite, occur in nature and might be stable in a bedded salt type of repository but in general no natural mineral of iodine hints of very long-term stability.

P.2.5.2 Framework Structures for Iodine

Two candidate minerals that are composed of three-dimensional frameworks contain cavities sufficiently large to house the I^- ion: sodalite and the boracite family.

Sodalite, $Na_4Al_3Si_3O_{12}Cl$, is a member of the feldspathoid group. It is a three-dimensional framework and the essential Cl^- is locked in cage-like interstices. Iodine can be substituted for Cl^- and maintained in this structure.

Boracite, $Mg_3B_7O_{13}Cl$ is a three-dimensional framework of B-O tetrahedra with the Cl^- locked in cage structure. Other minerals of the boracite family are ericaite, $(Fe,Mn)_3B_7O_{13}Cl$, and chambersite, $Mn_3B_7O_{13}Cl$. However, a very large number of synthetic materials with the boracite structure have been synthesized. Many of the synthetics contain I^- rather than Cl^- . They are stable under hydrothermal conditions.

P.2.5.3 Lead Oxyhalides

There exists a small group of minerals composed of the oxy- or hydroxy-halides of lead. These materials usually appear as oxidation products on lead-zinc ores which is evidence for their stability in the surface environment. The list includes:

Murdochite	$PbCu_6(O,Cl,Br)_8$
Mendipite	$Pb_3Cl_2O_2$
Penfieldite	$Pb_2Cl_3(OH)$
Yedlinite	$Pb_6CrCl_6(O,OH)_8$
Phosgenite	$Pb_2(CO_3)Cl_2$

Little is known of the structures, solubilities, and ranges of stability of these materials. The substitution of iodine for chloride in the lead oxyhalide structures should be investigated.

P.2.6 Uranium Minerals

Uranium occurs in nature in both the U^{+4} and U^{+6} valence state. The U^{+5} valence state has been postulated, especially in U_3O_8 and other oxides intermediate between UO_2 and UO_3 , but it has not really been verified. Its existence is not critical to our discussion.

P.2.6.1 U^{+4} Minerals

Uranium occurs as U^{+4} in only a small group of minerals. The most important and best known is uraninite, UO_2 , which has the fluorite, CaF_2 , structure. It is the principal mineral in most uranium deposits and is found in pegmatites, in sandstones and metasediments, and as an accessory mineral in some granites. Natural UO_2 is rarely stoichiometric and is better described as UO_{2+x} where x ranges between 0 and 0.25. Most uraninite from older sources is metamict and may be called pitchblende.

In sandstone deposits the uraninite has formed from circulating ground water by reduction of the U^{+6} . In the reduced form it is very stable and is common in the placer deposits of the Witwatersrand district in Africa. These uraninite grains were carried down streams and deposited in energetic depositional environments without chemical breakdown because the atmospheric conditions of the time were highly reducing. If uraninite could be maintained in its U^{+4} state it would be a good repository mineral. Unfortunately, it alters rapidly in present-day atmospheres.

Uraninite is usually only uranium bearing in sandstone deposits, but in pegmatites it may contain significant quantities of Ce and Th in solid solution. Actually, complete solid solutions of these elements can be prepared under laboratory conditions.

Some of the other U^{+4} minerals occur in quantities sufficient for them to be called ore minerals. Coffinite, $USiO_4$, brannerite, UTi_2O_6 , and ningyoite, $CaU(PO_4)_2 \cdot 1.5H_2O$, occur primarily in sedimentary or metasedimentary environments probably as syngenetic minerals. Other U^{+4} minerals include lermontovite, $(U,Ca,Ce...)_3(PO_4)_4 \cdot 6H_2O$; sedovite, $U(MoO_4)_2$; uranopyrochlore, $U_2Nb_2O_6(O,OH,F)$; cliffordite, UTe_3O_8 , and ishakowite, $(U...)(Nb,Ta)O_4$. In addition U^{+4} occurs as a minor element in many minerals, mostly replacing other group IV elements or the rare earths. At the conditions existing at the earth's surface all these U^{+4} minerals readily alter by oxidation and weather by releasing the uranium into the ground water system. The U^{+6} may be fixed immediately in new minerals or may migrate for long distances before being redeposited.

P.2.6.2 Uranate Minerals

Uranium as U^{+6} forms a large group of oxides, hydrated oxides, and uranates. The uranates form compounds with Na, K, Mg, Ca, Ba and Pb. Some of these compounds are anhydrous, but most are hydrates. There are many crystalline modifications of UO_3 but none occurs naturally. Usually the hydrate schoepite, $UO_3 \cdot 2H_2O$, or one of its polymorphic forms occurs. If the other elements are present the tendency is to form the uranate minerals.

The uranates occur in the immediate vicinity of the source mineral, usually uraninite. They develop as a replacement aureole of poorly crystallized phases commonly called gummite. The Pb which is common in older deposits is primarily radiogenetic in origin.

The uranates do not survive further weathering and are replaced by uranyl compounds in the main oxidized zone of any ore body. It is doubtful if any uranate would be a good uranium repository.

P.2.6.3 Uranyl Minerals

Any uranium which finds its way into the ground water system migrates as the uranyl ion, UO_2^{+2} , or as some complex involving the uranyl ion. As the local conditions change the uranyl ion may precipitate as one of over 100 mineral species.

P.2.6.4 Uranyl Ion

The uranyl ion is a linear group with the uranium in the center and the oxygen ions on the ends. Because of this unique geometry uranyl compounds form their own series of compounds in nature with very little substitution of other ions.

Uranyl will form complex structures with almost any oxyanion, carbonate, sulfate, phosphate, arsenate, molybdate, selenate, vanadate and silicate. The crystal structure of the minerals is usually uranyl-oxyanion sheets or chains, which stack so as to contain interstitial low-charge cations and water molecules. Most of the carbonates, sulfates, molybdates and selenates and even the silicates are moderately soluble and will leach as the environmental conditions change. The phosphates-arsenates and vanadates appear to be very insoluble and may be potential repository compounds. The known minerals are listed in Table P.2.4.

The uranyl phosphates and arsenates are usually considered together because their crystal chemistry is very similar and in some cases there is even partial substitution of phosphorus and arsenic. In all compounds these ions exist in tetrahedral coordination. Vanadium is tetrahedral in a few vanadates, but in most vanadates complex V_2O_8 groups of pentagonal edge-shared VO_5 coordination polyhedra are formed.

As can be seen in Table P.2.4, the phosphates-arsenates-vanadates are usually classified by their U:X ratio where X is P, As, V. Several ratios exist but the most common is the U:X = 1. Within this group are several minerals that have great potential as repository minerals. This potential is suggested by the wide range of occurrence, the frequency of mineral formation and the extremely low solubility of the compounds.

TABLE P.2.4. Uranyl Phosphates, Arsenates, Vanadates

$UO_2:XO_4$		
4:2	Arsenuranylite	$Ca(UO_2)_4(AsO_4)_2(OH)_4 \cdot 6H_2O$
	Bergenite	$Ba(UO_2)_4(PO_4)_2(OH)_4 \cdot 8H_2O$
	Renardite	$Pb(UO_2)_4(PO_4)_2(OH)_4 \cdot 7H_2O$
3:2	Troegerite	$(UO_2)_3(AsO_4)_2 \cdot 12H_2O$ (see 2:2)
	Huegelite	$Pb_2(UO_2)_3(AsO_4)_2(OH)_4 \cdot 3H_2O$
	Dumontite	$Pb_2(UO_2)_3(PO_4)_2(OH)_4 \cdot 3H_2O$
	Phosphuranylite	$Ca(UO_2)_3(PO_4)_2(OH)_4 \cdot 7H_2O$
2:2	Carnotite	$K_2(UO_2)_2(V_2O_8) \cdot 3-5H_2O$
	Tyuyamunite	$Ca(UO_2)_2(V_2O_8) \cdot 5-8H_2O$
	Metatyuyamunite	$Ca(UO_2)_2(V_2O_8) \cdot 3H_2O$
	Curienite	$Pb(UO_2)_2(V_2O_8) \cdot 5H_2O$
	Francevillite	$Ba(UO_2)_2(V_2O_8) \cdot 5H_2O$
	Strelkinite	$Na_2(UO_2)_2(V_2O_8) \cdot 6H_2O$
	Autunite	$Ca_{1-2}(UO_2)_2(PO_4)_2 \cdot 8-12H_2O$
	Meta-autunite I	$Ca_{1-2}(UO_2)_2(PO_4)_2 \cdot 6-8H_2O$
	Meta-autunite II	$Ca_{1-2}(UO_2)_2(PO_4)_2 \cdot 4-6H_2O$
	Meta-vanuralite	$Al(UO_2)_2(VO_4)_2(OH) \cdot 8H_2O$
	Vanuralite	$Al(UO_2)_2(VO_4)_2(OH) \cdot 11H_2O$
	Vanuranylite	$(H_3O, Ba, Ca, K)_{1.6}(UO_2)_2(VO_4)_2 \cdot 4H_2O$
	Dewindtite	$Pb(UO_2)_2(PO_4)_2 \cdot 3H_2O$
	Sengierite	$Cu(UO_2)_2(V_2O_8) \cdot 8-10H_2O$
	2:3	Coconinoite
2:4	Parsonsite	$Pb_4(UO_2)_2(PO_4)_2 \cdot 2H_2O$
	Przhevalskite	$Pb(UO_2)_2(PO_4)_4 \cdot 4H_2O$
	Pseudoautunite	$(H_3O)_4Ca_2(UO_2)_2(PO_4)_4 \cdot 5H_2O$
	Walpurgite	$(BiO)_4(UO_2)_2(AsO_4)_4 \cdot 6H_2O$
	Hallimondite	$Pb_2(UO_2)_2(AsO_4)_2$

The abundance of uranyl phosphates and arsenates results more from the stability of uranyl phosphate and uranyl arsenate complexes in ground water (Langmuir 1978) than from any abundance of P or As. The complex polymerizes readily into sheet-like crystal structures, which incorporate a variety of low-charge cations and water molecules between the sheets. Thus, they form a large number of mineral species depending on the available cation. The toxicity of As, however makes it less desirable additive.

The most important mineral family in the phosphates is the autunite minerals. The family is usually broken into three groups--autunite, meta-autunite I, and meta-autunite II,

depending on the number of water molecules involved. Table P.2.5 lists all the members of the autunite family. The variation of water is common to the group but does not seem to affect the stability of the species.

Autunites are known to form compounds with Ca, Mg, Ba, Na, Cu, Fe^{2+} , K, Zn, Mn, Co, Pb, NH_4 , Al, and H_2O . Many synthetic analogs can also be easily formed including Sr and even Li. The included cation is easily exchangeable in acid solutions but the autunite structure remains unaffected by the many substitutions.

TABLE P.2.5. The Autunite Family

Autunites, $\text{R}_{1-2}(\text{UO}_2)_2(\text{XO}_4)_2 \cdot 8-12\text{H}_2\text{O}$		
Autunite, $\text{Ca}(\text{UO}_2)_2(\text{PO}_4)_2 \cdot 8-12\text{H}_2\text{O}$		
Fritschelite	Mn	V
Heinrichite	Ba	As
Kahlerite	Fe	As
Novacekite	Mg	As
Sabugalite	H, Al	P
Saleeite	Mg	P
Sodium autunite	Na, Ca	P
Torbernite	Cu	P
Uranocircite	Ba	P
Uranospinite	Ca	As
Zeunerite	Cu	As
Meta-autunites, $\text{R}_{1-2}(\text{UO}_2)_2(\text{RO}_4)_2 \cdot 6-8\text{H}_2\text{O}$		
Abernathylite	$\text{K}_2(\text{UO}_2)_2(\text{AsO}_4)_2 \cdot 6-8\text{H}_2\text{O}$	
Bassetite	Fe^{2+}	P
Meta-ankoleite	K_2	P
Meta-autunite I	Ca	P
Metaheinrichite	Ba	As
Metakahlerite	Fe^{2+}	As
Metakirchleimerite	Co	As
Metalodevite	Zn	As
Metanovacekite	Mg	As
Metaforbernite	Cu	P
Meta-uranocircite	Ba	P
Meta-uranospinite	Ca	As
Metazeunerite	Cu	As
Sodium uranospinite	NaCa	As
Troegerite	$(\text{H}_3\text{O})_2(\text{UO}_2)_2(\text{AsO}_4)_2 \cdot 6\text{H}_2\text{O}$	
Uramphite	NH_4	P
unnamed	$(\text{H}_3\text{O})_2$	P
Meta-autunite II	$\text{Ca}(\text{UO}_2)_2(\text{PO}_4)_2 \cdot 4-6\text{H}_2\text{O}$	

In nature, autunite, $\text{Ca}(\text{UO}_2)_2(\text{PO}_4)_2 \cdot 8-12\text{H}_2\text{O}$, and meta-autunite I, $\text{Ca}(\text{UO}_2)_2(\text{PO}_4)_2 \cdot 6-8\text{H}_2\text{O}$, are very common anywhere uranium is found. They are found as a secondary mineral in all climates; and have been mined as ore minerals in several locations because of their abundance. In Cameron, Arizona, they occur in near-surface sandstone lenses and around Shoshoni, Wyoming. They are mined from bentonite pits where they form in the desiccation cracks of the clay. At Ningyo Prefecture in Japan they are found in sandstone, where they were mined extensively until the primary ningyoite zone was encountered. Some very noted specimen localities include the Daybreak Mine in Washington, and Cornwall, England. They are also common alteration products in uranium-bearing pegmatites. In all these localities they have proven to be very stable. The leaching characteristics under various conditions still must be tested.

Among the other uranyl phosphates several other candidates are also evident as possible repositories. In particular we should consider the phosphyranylite $\text{Ca}(\text{UO}_2)_4(\text{PO}_4)_2(\text{OH})_4 \cdot 7\text{H}_2\text{O}$. It is a much rarer mineral than autunite but has a higher loading factor because the U:P ratio is 3:2. Considerably less is known about the stability of this phase. Its conditions of formation and synthesis are less well known but it occurs similarly to autunite.

One must not overlook the vanadates as potential repository minerals, in particular carnotite, $\text{K}_2(\text{UO}_2)_2\text{V}_2\text{O}_8 \cdot 3-5\text{H}_2\text{O}$; tyuyamunite, $\text{Ca}(\text{UO}_2)_2\text{V}_2\text{O}_8 \cdot 5-8\text{H}_2\text{O}$, and metatyuyamunite, $\text{Ca}(\text{UO}_2)_2\text{V}_2\text{O}_8 \cdot 3\text{H}_2\text{O}$. These three minerals occur extensively throughout the Colorado Plateau and have been mined for uranium. They usually occur in sandstone lenses and are found in intersticed among the sand grains. Once formed, they appear to resist weathering and alteration even at surface conditions. Strontium analogs might easily be made. Ion exchange, common in the autunites, does not seem to occur in the vanadates.

P.2.7 Technetium Hosts

Since the element technetium is not known in nature, it follows that no minerals exist with technetium as an essential element. Technetium exists mainly in valence states Tc^{4+} and Tc^{7+} with the latter forming the very soluble pertechnetate ion. Technetium⁴⁺ forms stable solid oxide phases and, because of a similar ionic radius, behaves much like Ti^{4+} . Many titanium analogs have been synthesized (Muller et al. 1964) including spinels, pyrochlores, perovskites, and a stable solid solution between TiO_2 and TcO_2 . Titanium minerals may be the best hosts for technetium if reducing conditions are maintained in the repository.

P.3 MINERAL TABLES

P.3.1 Hosts for Radionuclides

Table P.3.1 lists selected minerals which have potential as hosts for radionuclides. The entries in Table P.3.1 were selected according to the criteria listed below.

Approximately 2500 mineral species have been identified. These have been compiled into reference sources of which those of the Dana system (Palache et al. 1944, 1951), Deer, Howie and Zussman (1962), Strunz (1970), and Roberts, Rapp and Weber (1974) were consulted. Each of the 2500 minerals was reviewed and in a first sieving all minerals that were known to be water-soluble, chemically undesirable or crystal-chemically unsuitable as radionuclide hosts were eliminated. A much shortened list of about 100 minerals remained. A second sieving eliminated minerals of great chemical complexity that would be difficult to synthesize. The minerals that remained were separated according to the radionuclide for which they were to serve as host and these groups were then roughly ranked with the best candidates listed first.

Table P.3.2 is the final listing. In addition to mineral name and formula, the table lists some available information of the occurrence of these minerals in nature, which provides clues to their stability in the repository environment, and on alteration processes where known. It must be emphasized that the data on these later categories are very sparse although this study does not claim to be an exhaustive literature survey. Table P.3.2 is intended as a guide for future research rather than finalized data for engineering design.

P.3.2 Commentary on Table P.3.1

The lack of silicate minerals on the listing is perhaps unexpected. Silicates make up the bulk of the rocks on the earth and many of them are very stable. However, the common silicate structures utilize the most abundant elements of the earth and the critical radionuclides from nuclear waste are, with the exception of ^{90}Sr , unusual elements, either too large or too small to fit into available sites in the silicate minerals. Furthermore, silicates are relatively less resistant to weathering and only a few, of which zircon is an outstanding example, survive the weathering process to become detrital minerals. Even fewer survive to become placer minerals.

Phosphates and oxides are the first and second most stable minerals in a wide variety of geochemical environments from initial formation at high temperatures and pressures, through weathering transport, contact with salt water in oceanic depositional basins, burial, diagenesis, upheaval, and in some cases a complete second cycle of weathering.

A very large number of phases on the list occur in pegmatites or in alkaline rocks that are closely related. The minerals, by implication, are stable in the presence of aqueous solutions at temperatures to 600°C and pressures to several kilobars. Chemical compatibility with granite rocks is implied. Whether many of these minerals are compatible with other candidate repository rocks, basalts, and shales require research. The fact that the minerals do not occur in these rocks in nature means only that the chemistry for their formation was not correct, not that the minerals are necessarily incompatible.

TABLE P.3.1. Selected Host Minerals for Radionuclides

Element	Host Mineral	Formula	Substitution(a)	Occurrence in Nature	Alteration
Cs	pollucite	$Cs_{2-x}Na_xAl_2Si_4O_{12} \cdot H_2O$	E	granite pegmatites	
Sr	anorthite (feldspar)	$Ca_{1-x}Na_xAl_{2-x}Si_{2+x}O_8$	R	basalt	slow break-down into clay minerals under surface weathering conditions
	Sr-apatite	$Sr_5(PO_4)_3(OH,F)$	E	alkalic pegmatites	
	belovite	$(Sr,Ce,Na,Ca)_5(PO_4)_3(O,OH)$	SS	alkalic pegmatites	breakdown at low pH
	celestine	$SrSO_4$	E	oxidation zones in sulfur deposits primary precipitation	
	Sr-autunite	$Sr(UO_2)_2(PO_4)_2$	E	strata-bound ore deposits	
	goyazite	$SrAl_3(PO_4)_2(OH)_5H_2O$	E	pegmatite	
	lamprophyllite	$Na_2(Sr,Ba)_2Ti_3(SiO_4(OH,F))_2$	SS	nepheline syenites alkali-rich pegmatites	
	lusangite	$(Sr,Pb)Fe_3(PO_4)_2(OH)_5 \cdot H_2O$	SS	pegmatite	
	bogildite	$Na_2Sr_2Al_2PO_4F_9$	E	cryolite	deposits
	danburite	$CaB_2Si_2O_8$	R	andesite	xenoliths
	attakolite	$(Ca,Mn,Sr)_3Al_6(PO_4,SO_4)_7 \cdot 3H_2O$	SS		
	cuspidine	$Ca_4Si_2O_7(F,OH)_2$	R	metamorphic rocks limestone contact zones	
	rankinite	$Ca_3Si_2O_7$	R	strain zones	gelatinizes readily at low pH
	melilite	$Ca_2Mg_{1-x}Al_{2x}Si_{2-x}O_7$	R	extrusive rocks	
	umbozerite	$Na_3Sr_4ThSi_8(O,OH)_{24}$	E		
	scheelite	$CaWO_4$	R	pegmatites	
	powellite	$CaMoO_4$	R	pegmatites	

TABLE P.3.1. (continued)

Element	Host Mineral	Formula	Substitution(a)	Occurrence in Nature	Alteration
I	sodalite	$\text{Na}_8\text{Al}_6\text{Si}_6\text{O}_{24}\text{Cl}_2$	R	nepheline-syenite rocks	
	boracite	$\text{Mg}_3\text{B}_7\text{O}_{13}\text{Cl}$	R	salt domes and salt deposits	occurs in the "water insoluble" fractions of salt deposits
	ericaitite	$(\text{Fe},\text{Mn})_3\text{B}_7\text{O}_{13}\text{Cl}$	R		
	chambersite	$\text{Mn}_3\text{B}_7\text{O}_{13}\text{Cl}$	R		
	parahilgardite	$\text{Ca}_2\text{B}_5\text{O}_8\text{Cl}(\text{OH})_2$	R	salt domes	occurs in "water insoluble" fraction
	murdochite	$\text{PbCu}_6(\text{O},\text{Cl},\text{Br})_8$	R	oxidation zones of Pb-Zn deposits	
	mendipite	$\text{Pb}_3\text{Cl}_2\text{O}_2$	R		
	penfieldite	$\text{Pb}_2\text{Cl}_3(\text{OH})$	R		
	yedlinite	$\text{Pb}_6\text{CrCl}_6(\text{O},\text{OH})_8$	R		
	phosgenite	$\text{Pb}_2(\text{CO}_3)\text{Cl}_2$	R		
marshite	CuI	E	associated with copper ores	darkens on exposure to air	
	iodargyrite	AgI	E	secondary mineral in silver ores	
	miersite	$(\text{Ag},\text{Cu})\text{I}$	E	associated with copper ores	
Tc	perovskite	CaTiO_3	R	basic igneous	rocks
	calziritite	$\text{CaZr}_3\text{TiO}_9$	R	carbonatite	partially dissolves in low pH solutions
	yttrocrasite	$(\text{Y},\text{Th},\text{U},\text{Ca})_2\text{Ti}_4\text{O}_{11}$	R		

TABLE P.3.1. (continued)

Element	Host Mineral	Formula	Substitution(a)	Occurrence in Nature	Alteration
Tc	batisite	$\text{Na}_2\text{BaTi}_2\text{Si}_4\text{O}_{14}$	R	nepheline syenite	
	brannerite	$(\text{U,Ca,Ce})(\text{Ti,Fe})_2\text{O}_6$	R	hydrothermal mineral	
Lantha- nides, Acti- nides	monazite	$(\text{Ce,L a})\text{PO}_4$	E	granites, pegmatites, placers, hydrothermal deposits metamorphic rocks	extremely stable
	cheralite	$(\text{Ce,Ca,Th})(\text{P,Si})\text{O}_4$	SS		sometimes yellow crust of rhabdophanite
	xenotime	YPO_4	E	granites, pegmatites, placers, hydrothermal deposits, sandstones	very stable alters to churchite
	rhabdophanite	$(\text{Ce,L a})\text{PO}_4 \cdot \text{H}_2\text{O}$	E	alkali peg- matites, hydrothermal deposits, sandstones	very stable forms from monazite but dehydrates to monazite on prolonged storage
	brockite	$(\text{Ca,Th,Ce})\text{PO}_4 \cdot \text{H}_2\text{O}$	E		
	grayite	$(\text{Th,Pb,Ca})\text{PO}_4 \cdot \text{H}_2\text{O}$	E		
	churchite	$\text{YPO}_4 \cdot 2\text{H}_2\text{O}$	E	alkali massifs limonite ores	forms from zenotime
	zircon	ZrSiO_4	R	acid and alkali igneous rocks pegmatites, placers	metamict highly resis- tant to wea- thering
	baddeleyite	ZrO_2	R	carbonatites, gabbro, placers, basalts	highly stable

TABLE P.3.1. (continued)

Element	Host Mineral	Formula	Substitution ^(a)	Occurrence in Nature	Alteration
Lanthanides, Actinides	tacheranite	$(Zr,Ca,Ti)O_2$	R	alkali massifs	
	bazirite	$BaZrSi_3O_9$	R	granites	
	zirkelite	$Zr(Ca,Th,Ce)(Ti,Nb)_2O_7$	SS	magnetite deposits pyroxenites	
	thorite	$ThSiO_4$	E	greisens from granites	metamict
	huttonite	$ThSiO_4$	E	sands	alters to Y-bastnaesite
	thalenite	$Y_2Si_2O_7$	E	pegmagites	
	yttrialite	$(Y,Th)_2Si_2O_7$	E		
	throtveitite	$(Sc,Y)_2Si_2O_7$	E	pegmatites	
	bastnaesite	$(Ce,La)CO_3F$	E	hydrothermal deposits, pegmatites, granites	gradual alteration to lanthanite, rhabdophanite or cerianite
	cordylite	$Ba(Ce,La)_2(CO_3)_3F_2$	E	alkali syenite	
parisite	$Ce_2Ca(CO_3)_3F_2$	E	detrital, hydrothermal deposits, pegmatites, carbonate ore bodies	replaced by bastnaesite	
synchysite	$CaCe(CO_3)_2F$	E	alkali syenite pegmatite		
röntgenite	$Ce_3Ca_2(CO_3)_5F_3$	E	pegmatite		
cerianite	$(Ce,Th)O_2$	E	carbonates pegmatites		
davidite	$(Fe,La,Ce,U)_2(Ti,Fe)_5O_{12}$	SS	granites skarns, pegmatites, with vein minerals		
euxenite	$Y(Nb,Ti)_2(O,OH)_6$	E } E } E } SS }	pegmatites granites placers	can be altered but somewhat stable	
polycrase	$Y(Ti,Nb)_2(O,OH)_6$				
delorenzite	$Y(Ta,Nb)_2(O,OH)_6$				
fersmite	$(Ca,Ce)(Nb,Ti,Fe)_2(O,OH,F)_6$				

TABLE P.3.1. (continued)

Element	Host Mineral	Formula	Substitution ^(a)	Occurrence in Nature	Alteration
Lanthanide, Actinide	columbite	$(\text{Fe}, \text{Mn})(\text{Nb}, \text{Ta})_2\text{O}_6$	R	granites, pegmatites, quartz veins, greisen deposits, placers	very resistant to weathering
	tantalite	$(\text{Fe}, \text{Mn})(\text{Ta}, \text{Nb})_2\text{O}_6$	R		
	perovskite	CaTiO_3	R		
	loparite	$(\text{Na}, \text{Ce}, \text{Ca})(\text{Ti}, \text{Nb})\text{O}_3$	SS	basic igneous rocks	can be altered to metaloparite but retains lanthanides
	aeschynite	$(\text{Ce}, \text{Nd}, \text{Th}, \text{Y})(\text{Ti}, \text{Nb})_2\text{O}_6$	SS	alkali massifs pegmatites	usually weathers
	polymignyte	$(\text{Ca}, \text{Fe}, \text{Ce})(\text{Zr}, \text{Ti}, \text{Nb})_2\text{O}_6$	SS		
	sinicite	$(\text{Ce}, \text{Nd}, \text{Th}, \text{U})(\text{Ti}, \text{Nb})_2\text{O}_6$	SS		
	fergusonite	YNbO_4	E	granitoid formations, placers, granites, pegmatites	fairly stable often associated with monazite in placers
	formanite	YTaO_4	E		
	samarskite	$(\text{Fe}, \text{Y}, \text{U}, \text{.})(\text{Nb}, \text{Ti}, \text{Ta})_2\text{O}_7$	SS	pegmatites, gold placers	
	pyrochlore	$(\text{Na}, \text{Ca}, \text{U}, \text{Ce}, \text{Y})_{2-x}(\text{Nb}, \text{Ta}, \text{Ti})_{206-x}(\text{OH}, \text{F})_{1+x}$	SS		
	betafite	$(\text{U}, \text{Ca}_{2-x}(\text{Nb}, \text{Ti}, \text{Ta})_2\text{O}_{6-x}(\text{OH})_{1+x}$	SS		
	zirconolite	$\text{CaZrTi}_2\text{O}_7$	R	Alkali rock massifs	pervasively altered
	microlite	$(\text{Ca}, \text{Na})_2\text{Ta}_2\text{O}_6(\text{O}, \text{OH}, \text{F})$	R		
	obruchevite	$(\text{Y}, \text{U}, \text{Ca})_{2-x}\text{Nb}_2\text{O}_6(\text{OH})$	SS		
	djalmaite	$(\text{Ca}, \text{Na}, \text{U})_2\text{Ta}_2\text{O}_6(\text{O}, \text{OH}, \text{F})$	SS		
	pandaite	$(\text{Ba}, \text{Sr})_{2-x}(\text{Nb}, \text{Ti})_{207-x} \cdot 7\text{H}_2\text{O}$	R		

TABLE P.3.1. (continued)

Element	Host Mineral	Formula	Substitution ^(a)	Occurrence in Nature	Alteration
U	uraninite	UO ₂	E	pegmatites	rapid in oxidizing conditions but very stable in reducing conditions
	carnotite	K ₂ (UO ₂) ₂ (VO ₄) ₂	E }	sandstone	relatively insoluble
	tyuyamunite	Ca(UO ₂) ₂ (VO ₄) ₂	E }		
	autunite	Ca(UO ₂) ₂ (PO ₄) ₂	E }	pegmatites, sandstone, sedimentary breccia	insoluble
	K-autunite	K ₂ (UO ₂) ₂ (PO ₄) ₂	E }		
	Sr-autunite	Sr(UO ₂) ₂ (PO ₄) ₂	E }		
	phosphuranylite	Ca(UO ₂) ₄ (PO ₄) ₂ (OH) ₄ 7H ₂ O	E	U-schists, pegmatites	
	ningyoite	(U,Ca,Ce) ₂ (PO ₄) ₂ 1-2H ₂ O	E	sedimentary rocks	
	lermontovite	(U,Ca,Ce) ₃ (PO ₄) ₄ 6H ₂ O	E		
	coffinite	U(SiO ₄) _{1-x} (OH) _{4x}	E	sandstone, sedimentary breccia, U-schists	
	ekaniite	(Th,U)(Ca,Fe,Pb) ₂ Si ₈ O ₂₀	SS	pegmatite veins	
	weeksite	K ₂ (UO ₂) ₂ Si ₆ O ₁₅ 4H ₂ O	E		
	soddyite	(UO ₂) ₅ Si ₂ O ₉ 6H ₂ O	E	pegmatite	

(a) Substitution of radionuclide into host mineral: E = essential element; SS = solid solution; R = replacement of another element by radionuclide.

TABLE P.3.2. Systematic Tabulations of Metamict Minerals^(a)

<u>SIMPLE OXIDES</u>	<u>Nb-Ta-Ti OXIDES</u> (A = U, Th, REE, Co, Na, K) Mg, Mn, Fe ⁺² , Pb = Nb, Ta, Ti, Fe ⁺³ , W)
*Uraninite (UO ₂)(b,c)	
*Rutile (TiO ₂)	ABO ₃ (Perovskite structure)
<u>PHOSPHATES</u>	
*Monazite (Conybeare and Ferguson 1948, Brooker and Nufield 1950)	*Loparite Irinite *Knopite
*Xenotime (Sidorenko 1963)	A _{2-x} B ₂ O ₇₋₃ nH ₂ O (Pyrochlore structure)
*Griphite (Peacor and Simmons 1972)	*Pyrochlore Betafite *Microlite Djalmaite Obruchevite *Zirconolite
<u>SILICATES</u>	
<u>Nesosilicates (Si:O = 1:4)</u>	
*Zircon	
*Thorite	A ₂ B ₅ O ₁₅ (Davidite structure) *Davidite
*Coffinite	
*Titanite (Higgins and Ribbe 1976)	ABO ₄ (Fergusonite structure) Formanite *Fergusonite Risorite
*Huttonite	
*Steenstrupine-Cerite	
*Britholite group	AB ₂ O ₆ (Columbite structure) *Columbite (Hutton 1959, Ewing 1976b)
*Lessingite Karnasurtite Karnocerite Tritomite Spencite Rowlandite Gadolinite	AB ₂ O ₆ (Euxenite structure) *Euxenite (Ewing 1976a) Polycrase Delorenzite *Fersmite
<u>Sorosilicate (Si:O = 2:7)</u>	
Thortveitite group	AB ₂ O ₆ (Priorite) *Priorite *Aeschynite *Bloomstrandine Polymignite
*Thalenite Yttrialite *Hellandite *Rincolite Epidote group *Allanite *Chevkinite *Perrierite	AB ₂ O ₄ (Samarskite structure) Samarskite Chlopinite Loranskite Yttrocrasite
*Vesuvianite (Bouska 1970)	
<u>Cyclosilicates (Si:O = 1.3)</u>	
*Eudialyte Cappelenite (Faessler 1942)	AB ₂ O ₆ (Brannerite structure) *Brannerite Thorutile AB ₂ O ₇ (Zirkelite structure) Zirkelite

(a) After Bouska (1970).

(b) The asterisk (*) indicates that the mineral also occurs as a partially or completely crystalline phase.

(c) A reference indicates that inclusion of the mineral in this table is based only on a single or poorly documented occurrence.

The rankings, except for the top few entries, are almost arbitrary. Although available mineralogical evidence suggests that these minerals are stable in the temperature and pressure regimes generally thought to exist around nuclear waste repositories, their relative stabilities are not known. Likewise, the relative solubilities of these generally insoluble phases are not known. Thus, detailed ranking or the construction of any sort of figure of merit cannot be done under the present state of knowledge.

Many of the oxide minerals are highly stable and insoluble because of a particular oxidation state. Lower oxidation states of the transition metals and of uranium form less soluble compounds than do the high oxidation states. The state of oxidation in a repository will be controlled by the oxidation potential and oxygen buffer capacity of the host rocks since these will be present in vastly larger volumes than the volume of the waste. Likewise the solubilities of many of the minerals are a sensitive function of the acidity of any circulating solutions. The fluorocarbonates are an example of minerals with low solubilities in neutral or alkaline solutions that become progressively more soluble as the pH decreases. The host rock in which the repository is formed will play an important role in buffering the oxidation potential and acidity of any circulating ground water that might contact the synthetic minerals of the waste form.

The large number of minerals that are listed as occurring in pegmatites is to be expected. Pegmatites are complex mineral assemblages that form from a residual high-water content fluid that remains after the crystallization of granitic rocks. Ions that are too big or too small or have the wrong charge or the wrong electronic structure to fit into any of the common granite minerals--quartz, feldspars, micas, and amphiboles, are concentrated in the residual fluid and finally crystallize into pegmatites. It is not the pegmatite-forming temperature and pressure regime that is critical but rather the complex solution chemistry that allows these minerals to be formed. Many of these minerals can be synthesized by entirely different methods but their occurrence in pegmatites does imply a substantial degree of mutual compatibility among the phases.

P.4 METAMICTIZATION

Metamict minerals are a special class of amorphous materials which were initially crystalline (Broegger 1893). Although the mechanism for the transition is not clearly understood, radiation damage caused by alpha particles and recoil nuclei is certainly critical to the process (Graham and Throber 1974, Ewing 1975). The study of metamictization of naturally occurring materials allows for the evaluation of the long-term effects that result from this type of radiation damage, particularly changes in physical properties. Comparison of metamict and non-metamict crystalline phases addresses the question of the susceptibility of different bonding and structure types to radiation damage and provides useful insights into defining radiation damage experiments.

P.4.1 Properties

The list below is an amplified tabulation of metamict mineral properties listed by Pabst (1952).

1. They are generally optically isotropic but may show varying degrees of anisotropy. Reconstitution of birefringence with heating is common.
2. Metamict phases lack cleavage. Conchoidal fracture is characteristic.
3. Some mineral species are pyronomic, that is, they glow incandescently on heating. In many cases, however, recrystallization may occur without observable glowing.
4. Crystalline structure is reconstituted by heating. The metamict material recrystallizes to a polycrystalline aggregate with a concomitant increased resistance to attack by acid. During recrystallization several phases may form, the particular phase assemblage is dependent on the conditions of recrystallization (e.g., temperature and type of atmosphere). In many cases the original pre-metamict phase may not recrystallize due to compositional changes caused by post-metamict alteration.
5. Metamict minerals contain U and Th, although contents may be quite variable (as low as 0.41% ThO₂ in gadolinite from Ytterby, Norway). Rare-earth elements are also common (in some cases over 50 wt%). Water of hydration may be high (up to 70 mole%).
6. They are x-ray amorphous. Partially crystalline metamict minerals display distinct line broadening and decreased line intensities. A shift of lines to lower values of two-theta is observed in specimens with a reduced specific gravity.
7. Some phases occur in both the crystalline and metamict state, and in these cases there is little chemical difference.

The most common methods of analysis of the metamict state are x-ray diffraction analysis of annealed material (Berman 1955, Lima-de-Faria 1964, Mitchell 1972, Ueda and

Koreskawa 1954) and differential thermal analysis (DTA) (Kerr and Hollan 1951, Orcel 1953, Kurath 1957). Most of the effort by mineralogists has been directed at establishing identification criteria.

Elemental analysis is commonly completed by wet chemical means on mineral separates or by standard electron microprobe analysis. The presence of water, both structural and absorbed, and the preponderance of rare-earth elements make a complete chemical analysis a rarity in the literature.

Although radiation damage experiments are voluminous, there have been only limited and unsuccessful efforts to simulate the process of metamictization under laboratory conditions (Mugge 1922, Primak 1954).

P.4.2 Summary of Observed Metamict Phases

To understand the compositional and structural controls on the process of metamictization, it is useful to tabulate naturally occurring metamict phases. Table P.3.2 listed those phases described as being partially or completely metamict in a review of the literature by Bouska (1970). This tabulation lists only the major compositional end-member. As one might expect for mineral groups of complex compositions (e.g., compare the A:B ratios for fergusonite and samarskite) that are metamict and much altered, the nomenclature of any single mineral group is quite complicated and much confused by the proliferation of varietal names (Ewing 1976). For a more detailed listing and discussion of the mineralogical literature the reader is referred to Bouska (1970).

The asterisk by each mineral name indicates it also occurs as a partially or completely crystalline phase. In some cases (e.g., monazite, xenotime and vesuvianite) the inclusion of a mineral phases as metamict is based only on a single or poorly documented occurrence. In these instances the critical reference is indicated. In other cases (e.g., rutile) the radiation damage was not caused by constituent uranium and thorium nuclides but rather occurred only along grain boundaries where the rutile was in epitaxial contact with radioactive davidite.

The uranium and thorium contents of phases that occur in both crystalline and metamict forms are interesting. Table P.4.1 gives the average U_3O_8 and ThO_2 contents of orthorhombic AB_2O -type Nb-Ta-Ti oxides. Although the data in the literature are limited, in general those specimens of euxenite, fersmite, aeschynite and lyndochite found in the crystalline state have distinctly lower uranium and thorium contents than their metamict euxenite and aeschynite counterparts. A similar relation has been demonstrated for zircons (Holland and Gottfried 1955, Krasnobayev et al. 1974).

Table P.4.2 is a compilation of radioactive minerals which are said to be always crystalline. Comparison of Tables P.3.2 and P.3.4 quickly reveals inconsistencies in the literature. Huttonite is listed as always crystalline (Pabst 1952) and partially metamict (Bouska 1970). Many of these inconsistencies may be resolved by very detailed and specific examinations of nomenclature. Also, note that among the phases listed as metamict (e.g., columbite and stitbiotantalite), their structures probably will not accommodate either

TABLE P.4.1. Uranium and Thorium Content (wt%) of Non-Metamict and Metamict AB_2O_6 --Type Nb-Ta-Ti Oxides

	<u>U₃O₈</u>	<u>ThO₂</u>
Non-metamict		
euxenite (Nefedor 1956)	(a)	(a)
fersmite (Alexandrov 1966)	(a)	(a)
aeschynite (Alexandrov 1962 and 1966)	not detected	0.72
allanite (Čech, Vrana and Povondra 1972)	0.25 ^(b)	2.26 ^(b)
lyndochite (Gorzhevskaya and Sidorenko 1962)	0.08 ^(c)	3.75
Metamict		
euxenite (mean value of 28 analyses)	9.31	3.08
aeschynites (mean value (of 22 analyses)	1.2	10.73

(a) Semiquantitative analysis, no U or Th reported.

(b) Analysis by R. C. Ewing, University of New Mexico.

(c) Reported as UO₃.

TABLE P.4.2 Radioactive Minerals Reported as Always Crystalline^(a)

autunite	$Ca(UO_2)_2 \cdot 10-12H_2O$
bastnaesite	$(Ce,La)(CO_3)F$
carnotite	$K_2(UO_2)_2(VO_4)_2 \cdot 3H_2O$
columbite	$(Fe,Mn)(Nb,Ta)_2O_6$
gummite	$UO_3 \cdot nH_2O$
huttonite ^(b)	$ThSiO_4$
metatorbernite	$Cu(UO_2)_2(PO_4)_2 \cdot 8H_2O$
monazite ^(b)	$(Ce,Th)PO_4$
stibiotantalite	$Sb(Ta,Nb)O_4$
thorianite ^(b)	ThO_2
thortveitite	$(Sc,Y)_2Si_2O_7$
tyuyamunite	$Ca(UO_2)_2(VO_4)_2 \cdot nH_2O$
uvanite	$U_2V_6O_{21} \cdot 15H_2O$
xenotime ^(b)	$(Y,U)PO_4$
yttriofluorite	Ca_3YF_9
titanite	$CaTiSiO_5$
uranite ^(b)	UO_2
baddeleyite ^(a)	ZrO_2

(a) After Ueda (1957).

(b) Primary phases which are invariably crystalline, even with high concentrations of uranium and thorium. Note that in some rare cases even these minerals have been reported as being partially metamict.

uranium or thorium. Reports of radioactive columbites are almost certainly mixtures of columbite and metamict microlite (Lima-de-Faria 1964). A number of the phases (bastnaesite and all hydrated phases) are alteration products. The primary phases that consistently occur in crystalline form, even with high concentrations of uranium or thorium, are indicated by asterisks.

P.4.3 Rate of Metamictization

The rate of metamictization of a given mineral to a first approximation, depends on: 1) the inherent stability of its structure and 2) the alpha particle flux resulting from the presence of uranium, thorium and their unstable daughter nuclides (Pabst 1952).

Pabst calculated that a minimum of 110,000 years is required for gadolinite, 0.4% Th, to become metamict. This figure, which could be low by a factor of 1000 (Ueda 1957, Lipova 1966, Hurley and Fairbain 1953), was obtained by assuming that all of that alpha decay energy was spent in disordering the structure and that this energy was measurable by DTA (Pabst 1952).

Most zircons become metamict upon receiving a radiation dose of about 10^{16} α /mg (Holland and Gottfried 1955). Using this dosage criterion, the following table gives estimates of the time required for some radioactive zircons to become metamict.

<u>Initial radionuclide content</u>	<u>Estimate time (yrs)</u>
1% Th	1.4×10^9
1% U	3.3×10^8
10% U	3.2×10^7
1% Pu ²³⁶ (does not exist in nature)	2.0

There are, however, zircons and thorites (thorite has the zircon structure and is expected to show similar radiation effects) which show anomalous radiation effects. Some zircons that have had radiation doses of only 2.8×10^{15} α /mg are metamict (Krasnobayev et al. 1974). On the opposite extreme is a report of a non-metamict thorite containing 10% uranium that is at least 1.2×10^8 years old (Hutton 1950). If this age is correct, then the thorite specimen has withstood a radiation dose of about 9×10^{16} α /mg. These data suggest that factors other than structural stability and alpha particle flux are important in determining the rate of metamictization.

P.4.4 Alteration Effects

Minerals that occur in the metamict state are often severely altered, either as a result of hydrothermal alteration or surface weathering. The resulting complicated compositional variates are in part responsible for the very complex mineral nomenclature. Most of the available data on alteration effects pertains to various Nb-Ta-Ti oxides (Ewing 1975, Wambeke 1970) and zircon, $(Zr,U)SiO_4$. In both cases alteration may be extensive and followed by recrystallization of phases quite different from the original pre-metamict phase (Ewing 1974).

For metamict, AB_2O_6 -type, Nb-Ta-Ti oxides (A = REE, Fe^{+2} , Mn, Ca, Th, U, Pb; B = Nb, Ta, Ti, Fe^{+3}) primary hydrothermal alteration causes a consistent increase in calcium content, generally a decrease in the uranium and thorium content, a decrease in total rare-earth concentrations, a slight decrease in B-site cations, and an increase in structural and absorbed water. Secondary alteration caused by weathering is similar in effect but produces a decrease in Ca content, an increased leaching of A-site cations and a relative increase in B-site cations. Refractive index, specific gravity and reflectance decrease with both types of alteration, but VHN_{50} remains approximately constant. It is important to note that although alteration effects in these natural materials have been carefully documented, there are no experimental data on hydrothermal alteration effects, solubility as a function of degree of metamictization, or the kinetics of these reactions.

There is an abundant literature on metamictization and alteration effects observed in zircon, $(Zr,U)SiO_4$, a phase commonly used by geologists in U/Pb radiometric dating. A summary of this literature is beyond the scope of this Appendix, but it should be the subject of future research. Discordant ages reported for metamict zircons indicate that the U/Pb ratios can be changed or slightly disturbed by alteration (Krogh and Davis 1975). Laboratory experiments involving zircon have demonstrated that altered regions are more rapidly dissolved by 48% hydrofluoric acid. There are some data which suggest that zircons that have become metamict are susceptible to attack by solutions that can cause alteration (Krogh and Davis 1975; Larsen et al. 1953). However, Mumpton and Roy (1961) have recrystallized numerous metamict zircons by hydrothermal treatment at temperatures of 500°C and above, and found that the Zr:Si ratio remained close to 1:1. This is an indication that neither element was selectively dissolved. They also demonstrated that the water often found in metamict zircons was molecular H_2O and not the result of H^+ ion exchange leaching. The data are still too limited to draw broad conclusions regarding the effect of metamictization on solubility, even for metamict zircons. Yet, at worst, this does not seem to be a major problem. Monazite, a mineral that apparently does not metamictize, was chosen as the lanthanide and actinide synthetic mineral in the reference scenario (see Section 3.2.1.3).

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FINAL

ENVIRONMENTAL IMPACT STATEMENT

**Management of
Commercially Generated
Radioactive Waste**

**Volume 3
Public Comments
Hearing Board Report**



October 1980

**U.S. Department of Energy
Assistant Secretary for Nuclear Energy
Office of Nuclear Waste Management
Washington, D.C. 20545**

VOLUME 3CONTENTS

INTRODUCTION	1
WRITTEN PUBLIC COMMENTS - POLICY ISSUES AND RESPONSES	5
WASTE PROGRAM	6
LICENSING AND THE DECISION-MAKING PROCESS	11
SITING ISSUES	15
FOREIGN WASTES	19
FUEL CYCLE ISSUES	20
LONG-TERM STORAGE	21
ATTRIBUTION OF COSTS/RISKS	22
ALTERNATIVE DISPOSAL ISSUES	23
GENERAL COMMENTS	24
WRITTEN PUBLIC COMMENTS - TECHNICAL ISSUES AND RESPONSES	25
ORGANIZATION AND PRESENTATION	26
SCOPE	34
RADIOLOGICAL ISSUES	39
CONSEQUENCE ANALYSIS	61
DOSE CALCULATIONS	113
RISK PERSPECTIVES	128
WASTE MANAGEMENT OPERATIONS	139
Reactor Operations	139
Waste Characterization	140
Waste Treatment and Packaging	144
Waste Storage	155
Waste Transportation	162
Decommissioning	167
Repository Construction and Operation	169
FUEL CYCLES	188
COSTS	190

SAFEGUARDS	199
GROWTH SCENARIOS	204
GEOLOGIC CONSIDERATIONS	206
MULTIBARRIERS FOR DISPOSAL	271
SOCIOECONOMIC/SOCIOPOLITICAL ISSUES	283
RESOURCE REQUIREMENTS	294
REFERENCE ENVIRONMENT	296
COMPARATIVE ASSESSMENT	300
ALTERNATIVE DISPOSAL CONCEPTS	313
General	313
Chemical Resynthesis	317
Very Deep Hole	320
Rock Melt	329
Island Disposal	338
Subseabed Disposal	351
Ice Sheet Disposal	377
Well Injection	383
Transmutation	394
Space Disposal	399
HEARING BOARD REPORT AND RESPONSES	408
REFERENCES	439
APPENDIX A: LIST OF RESPONDENTS TO DOE/EIS-0046D	A.1
APPENDIX B: INDEX FOR COMMENT LETTERS	B.1
APPENDIX C: FEDERAL AND STATE COMMENT LETTERS	C.1

INTRODUCTION

This volume is prepared in compliance with present regulations which provide for the assessment and consideration of public comment on draft environmental impact statements (40 CFR 1503.4, 1506.6). The purpose of this volume is to summarize public response to the draft Statement (DOE/EIS-0046D). Recommendations for revision of the draft Statement, provided through individual written comment and through the Hearing Board, are presented with a description of the way in which these comments were used during preparation of the final Statement.

Notice of availability of the draft Statement appeared in the Federal Register of April 20, 1979. This Federal Register notice requested interested public groups and individuals to review and comment on the draft Statement. Comments on the draft Statement came in the form of written comments addressed to the Department of Energy and/or as oral testimony given before the Hearing Board. Discussion of written comments comprise the major portion of this volume and are presented first. Discussion of the Hearing Board recommendations (which were based on the oral testimony) are presented following the written comments.

Written Comments

Written comments on the draft Statement were received from numerous sources, including individuals, representatives of industry, public interest groups, and state and Federal government agencies. Two hundred and nineteen comment letters were received. Individual comments presented in the letters ranged from one to approximately 300 for a total of greater than 2000 separate comments identified in the letters received by DOE.

Each letter was given an identification number when it was received. Letters were then examined and substantive comments were identified. As shown in Figure 1, comments were categorized into three general areas: policy, technical, and editorial. Policy issues are, in general, those which recommend an approach, courses of action or legalistic concerns regarding issues of waste management. Technical issues are those which specifically address the content, presentation and/or the analysis presented in the draft Statement: Policy and technical categories were further subdivided into topic areas. In the case of comments regarding policy, nine specific topic areas were identified. In the case of technical comments, there were eighteen topic areas. After comments were assembled into topic areas, they were gathered together in similar groups and paraphrased as summary or key issues where possible. Key issues were then addressed and responses were prepared for each key issue. In many cases, comments were of such a specific nature that they could not be grouped as summary or key issues within a topic area. In such instances, responses were prepared for the individual comment. Editorial comments (i.e., spelling and grammatical errors, incorrect cross-referencing, and errors in tables and figures) were considered during preparation of the final Statement and the appropriate changes were made. Such comments have not been discussed in this volume.

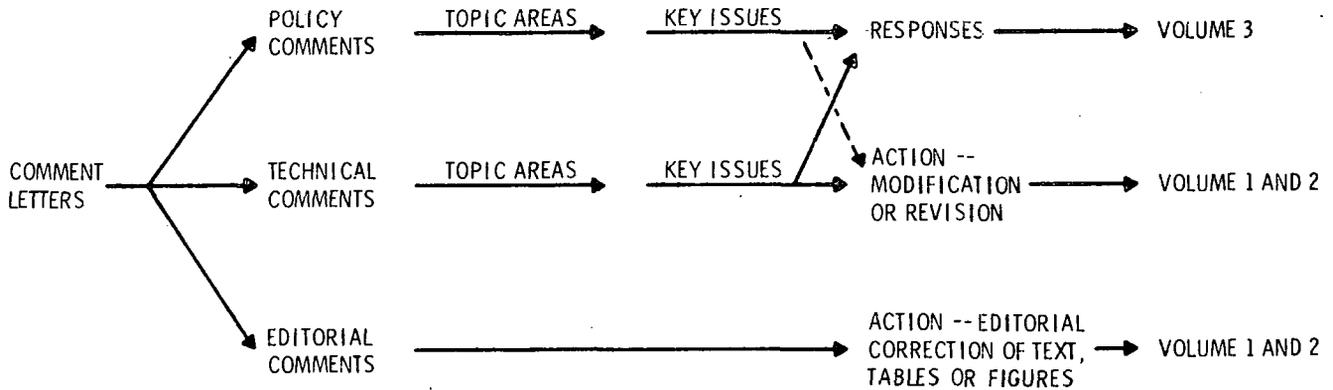


FIGURE 1. Process for Handling Public Comment Letters

This written comment section is structured such that policy comments (by topic area) are presented first followed by technical comments. Within a given topic area issues and responses are presented sequentially by page number of the draft Statement (where such page numbers were provided by the commenter). After each issue is stated the pertinent letter number(s) is cited in parentheses.

In addition, those issues raised by Federal Agencies (Department of Commerce - DOC, Department of Health, Education, and Welfare - HEW, Department of Interior - DOI, Environmental Protection Agency - EPA, Federal Energy Regulatory Commission - FERC, Nuclear Regulatory Commission - NRC) show the agency abbreviation next to the letter number.

The form of the response to each issue is as follows:

- a) Responses to policy issues will be either as a direct response to the issue or as a statement identifying why the issue is not considered to be within the scope of this Statement. In some cases, policy issues did result in changes to the statement and these are noted.
- b) In the case of technical issues, the response to the issue will identify how and to what degree the issue has been incorporated into the final Statement. Where possible the response will identify where in the final Statement the change was made. For technical comments addressing concerns outside the scope of the document, a statement is made to that effect.

Hearing Board Recommendations

This section of Volume 3 presents responses to the recommendations of the Hearing Board. To provide the reader with some background, this section contains a discussion on the Board and its members and contains a reproduction of the Hearing Board's report.

Three appendices appear at the end of the volume. Appendix A identifies those individuals and organizations that commented in writing on the draft Statement and identifies the number assigned to each letter. These numbers appear after each issue. Appendix B notes for each letter, the topic areas under policy and technical categories and the

particular page(s) of Volume 3 on which the letter has been cited. Appendix C contains reproductions of those letters submitted by Federal agencies and cover letter from states or state agencies.

As a result of the organizational changes made to the draft Statement, the tables below have been provided. These tables identify the correspondence between sections of the final Statement and the draft. These tables will be of assistance to the reader during the remainder of Volume 3 when reference is made to either the draft or final Statements.

Volume 1

<u>Section of Final Statement</u>	<u>Corresponding Section of Draft</u>
Foreword	Foreword
1.0	1.0
2.0	Foreword, 2.1
3.1	--
3.2	2.1
3.3	2.2, 2.3
3.4	Portions of 3.1.3 dealing with risk and risk perspectives
3.5	Portions of 3.1.2 and 3.1.3 dealing with non-technical issues
4.1	--
4.2	2.1
4.3	3.1.4, 3.2, portions of 3.9 and Appendix L
4.4	3.1.4 and portions of Appendix M
4.5	3.1.4 and portions of Appendix N
4.6	3.1.4 and portions of Appendix O
4.7	3.1.5.3
4.8	3.1.5.3
4.9	3.1.5.3
5.1	3.1.1 plus portions of 3.1.3 relating to the multiple barrier concept
5.2	3.1.2, 3.1.3, and 3.1.6
5.3	3.1.5.1
5.4	3.1.5.1
5.5	3.1.5.2
5.6	3.1.5.1
5.7	3.1.5.1
6.1	3.3-3.10
6.2	4.0
7.0	Portions of 3.1.5.4

Volume 2

<u>Section of Final Statement</u>	<u>Corresponding Section of Draft</u>
Appendix A	Appendix A
Appendix B	---
Appendix C	Appendix C
Appendix D	Appendix D
Appendix E	Appendix E
Appendix F	Appendix F
Appendix G	Appendix G
Appendix H	Appendix H
Appendix I	---
Appendix K	Appendix K, Appendix Q
Appendix L	---
Appendix M	---
Appendix N	---
Appendix P	Appendix P

WRITTEN PUBLIC COMMENTSPOLICY ISSUES AND RESPONSES

This section contains policy issues raised by the comment letters on the draft Statement as well as responses to these issues. For the purpose of this volume, policy issues are considered to be predominately of a subjective nature. In general, policy issues recommend an approach, courses of action or legalistic concerns regarding issues of waste management. In most cases, policy issues did not directly address the draft Statement. However, the Department does recognize the importance of such comments and has, therefore, provided responses to these comments.

WASTE PROGRAM

Issues

Several commenters pointed out that development of a geologic repository should place emphasis on:

- Demonstrating one repository medium while studying others because technology does exist to initiate an environmentally acceptable waste management program. (45, 154)
- Proceeding with investigations into a number of sites in a variety of media in different geographic regions either to ensure that the program is conducted in a technically conservative manner or to encourage timely implementation of a waste management program. (21, 28, 128, 166, 180)

Response

In his February 12, 1980 statement on radioactive waste management and disposal, the President called for the qualification of a number of alternative sites in a variety of media in different geographic regions.

Issues

Several commenters stated that the waste management program should be concerned about using resources (eg. chromium, copper and nickel) largely imported at this time because of potential economic or political pressures that could be brought to bear. (2, 30, 144) Other commenters noted that the isolation strategy chosen should assure protection of valuable/limited natural resources (eg. salt domes which may contain gas, oil, sulfur, potash in addition to salt). (22, 139, 218-DOI)

Response

In performing the environmental analysis of the total systems involved in waste disposal strategies, one element analyzed is the resources committed, such as land, water and materials. Part of the decision-making process required of agencies by the National Environmental Policy Act is a consideration of significant effects on natural resources, renewable and nonrenewable, including any mitigation measures which might be taken to lessen adverse impacts. This Statement provides such analysis of effects on resources in a generic sense; site or (program) specific statements would provide more detailed information.

One of the functions of a generic environmental impact statement is to indicate resource requirements for each alternative strategy under consideration. Should the selected program require that waste be placed in stainless steel canisters for disposal, consideration would be given in future environmental statements to this area of concern. Another of the functions of a generic environmental impact statement is to bring matters of key resource commitment to

WASTE PROGRAM

the attention of the decisionmakers, as has been done in this Statement (see final Sections 4.7, 5.4, 7.5). Further refinements in the estimates of use of resources will be developed for site specific statements.

Issue

Several commenters stated that program approaches and geologic disposal choices should require retrievability of wastes after emplacement (i.e., to take advantage of improvements in technology or discovery of a more acceptable site, to provide availability of access in case of malfunction, and to take advantage of the potential future resource value of the waste). (3, 27, 45, 96, 151, 164, 187, 196, 205)

Response

DOE defines retrievability as the removal of waste following emplacement in a repository but prior to closure of the repository system (backfilling of rooms and shafts). The initial emplacement of wastes so that they can be retrieved for periods of five and up to twenty-five years following placement, is a basis for the analysis of geologic repositories in this Statement. Further discussion of retrievability is in Section 5.3 and a description of requirements for five and twenty-five year ready retrievability is provided in Appendix K. The exact period of retrievability that will be required for a waste repository has not yet been determined.

Initial emplacement of waste in at least the first repository will permit retrievability for some initial period of time to allow for removal of emplaced wastes if unexpected phenomena are observed which could lead to the failure of the repository system to provide the required isolation or containment of the radioactive wastes.

Since the radioactive wastes are continually undergoing radioactive decay, the heat emission and radiation levels will be constantly decreasing. Consequently, the period immediately following waste emplacement will provide the most severe conditions on the nearfield environment. At the same time, accelerated tests will be performed at the repository level with artificially heated or concentrated waste forms, probably in closer arrays, to seek out any unpredicted phenomena. After an initial period of operations, much greater confidence in the understanding of the interaction between the waste and the repository structure will have been gained and permission to abandon the ability for ready retrieval will be sought from the NRC.

Issues

Several commenters expressed concern about the multiple barrier concept:

- More research and development is needed in the area of rock mechanics. (21)

WASTE PROGRAM

- The multibarrier approach must/should be adhered to in any disposal option, and waste forms and containers should receive equal attention to medium and disposal site. (28, 152)
- The multibarrier approach appears to be a false concept, because not enough is known about the systems involved or the interactions. (42, 68, 98)

Response

The Department of Energy's Statement of Position in the recent NRC rulemaking proceedings on storage and disposal of nuclear waste (DOE 1980a) contained the following quotation:

"In order to meet the primary objective for a waste disposal system, namely to isolate the wastes from the biosphere and to pose no significant threat to the public health and safety, waste must be prevented from reaching the human environment in quantities in excess of those permitted by radiation standards. . . The multibarrier concept requires that the success of the system be protected against deficient barrier performance or failure, by using a series of relatively independent and diverse barriers that would not be subject to common mode failure. Barrier multiplicity is required both as a hedge against unexpected occurrences or failures and to provide appropriate means for protecting against a wide variety of potentially disruptive events. Acceptable system performance must not be contingent on the performance of any non-independent barrier combinations."

DOE plans to continue to pursue the multibarrier approach to repository design recognizing that development of detailed identification of system components will be dictated by site-specific parameters.

Issue

Two commenters stated that the policy of not requiring the canister to provide containment after emplacement was a negligent one in that it would be difficult to control the wastes after emplacement. (9, 36)

Response

DOE anticipates that the canister will provide containment for a hundred years or more following emplacement. The final Statement has been changed to reflect this (see Section 5.1.2). It is uncertain at this time whether NRC will require that the canister function as a barrier to radionuclide migration for a specified period of time. See also response above.

Issue

One commenter suggested that if the material presented in the Statement is valid (that is, from a technical standpoint, no reason exists why the development of mined geologic repositories cannot proceed) then a large information gap exists between DOE and the public because a large segment of the public and many public officials believe this is the worst problem facing the U.S. (7)

WASTE PROGRAMResponse

The Report to the President by the Interagency Review Group (IRG) on Nuclear Waste Management, called for, "sustained, effective efforts to inform the public and to provide opportunities for discussion between the public and the government." These words appeared in the draft Report of the IRG when it was first released in October 1978 (IRG 1978). They were retained and even further emphasized in the final version of the report March 1979 (IRG 1979). DOE's plans for implementing this policy were announced May 1979, in the National Waste Terminal Storage Public Information Plan (DOE 1979a).

DOE believes that state and local governments, interest groups, and citizens in general must be informed about what is being considered, and that they must be involved in the decision-making process. DOE is seeking to implement this important recommendation of the Interagency Review Group.

To assist in carrying out these functions, DOE has developed a spectrum of activities which range from those normally associated with efforts to inform the public to those activities which are deliberately intended to facilitate the exchange of information between participants in the program and members of the public.

The preparation of this Statement and the recent DOE Position Paper to the NRC rule-making on nuclear waste storage and disposal (DOE 1980a) will hopefully bridge the "information gap" the commenter refers to.

Issue

Several commenters suggested that before DOE begins a long range program for a permanent disposal of radioactive waste, DOE should recognize (i.e., should consider as an issue in this statement) the potential value of the waste as a future resource and should make provisions to store this material in a retrievable mode, if necessary. (29, 35, 164, 187, 198, 205, 218-DOI)

Response

The primary goal in disposing of the radioactive material is to ensure its safe, permanent isolation from man's environment. To provide for the long-term retrievability of these wastes would potentially compromise the integrity of the repository system. If future decisions are made that the waste material is a valuable resource, the decision made may be to extract such material. This, however, is not a consideration in DOE's plans for the disposal of commercially-generated radioactive waste at this time.

Issue

Several commenters stated that the use of pilot scale facilities should be a part of the overall waste management program. (32, 152, 153, 203)

WASTE PROGRAMResponse

The objective of the program is to ensure the development of safe permanent isolation and disposal of radioactive wastes with a high degree of assurance in the safety and permanency of such isolation and disposal. The program is oriented to achieve this assurance as expeditiously as possible without a premature commitment to unproven solutions. If the use of pilot scale facilities is viewed as a way to enhance achievement of these objectives then such facilities would be utilized.

Issues

Numerous letters provided suggestions or stated preferences as to how the Department of Energy should conduct its waste disposal program(s). For instance, DOE should:

- Proceed with the demonstration/establishment of a waste repository. (3, 12, 13, 17, 21, 27, 32, 33, 34, 35, 49, 53, 55, 64, 152, 154, 166, 168, 177, 180, 182, 183, 196, 205, 210)
- Emphasize geologic disposal but continue studying other concepts. (32, 34, 180, 196)
- Move more deliberately and cautiously. (2, 28, 41, 86, 122, 128, 133, 153, 158, 164, 167, 187, 191, 195, 217, 218-D01)
- Continue testing alternatives. (28, 158, 199, 215, 218-D01)

Response

The recommended strategy supported by the analysis in this Statement and put forth by the Presidential Statement of February 12, 1980, is that several alternate sites for geologic disposal will be qualified in different geologic environments and host rock types prior to the selection of a preferred site for construction of a repository.

It is estimated that four or five such alternate sites could be qualified on the basis of exploratory and geophysical techniques from the surface by 1985 which could lead to an operating repository by the mid-1990's. If in-situ exploration at depth is required at each of these alternate sites before a license application for construction can be submitted to the Nuclear Regulatory Commission this schedule will slip by at least three years.

The strategy of delaying selection of the first repository site until several qualified alternatives are available is perceived as providing greater assurance of program success.

LICENSING AND THE DECISION-MAKING PROCESSIssue

One commenter stated that all nuclear waste repositories should be Federally licensed.
(3)

Response

DOE agrees with this statement as it applies to commercial high-level (HLW) and transuranic wastes (TRU). According to Federal law, all repositories for disposal of commercial waste and DOE repositories for high-level waste are subject to licensing by the Nuclear Regulatory Commission. Legislation has been proposed to Congress to extend the requirement to disposal of all transuranic wastes.

Issue

A commenter stated that states which have nuclear waste repositories should have state laws to govern health and safety at the repository, with the Federal government paying for state enforcement of those laws. (3)

Response

At this time, the most appropriate relationship between state and Federal governments concerning state jurisdiction and financial responsibilities over nuclear waste repository siting and monitoring is still being evaluated. Any state in which a repository might be located will most likely have specific authority for health and safety regulation which will be recognized by the Department of Energy.

Issues

Several commenters suggested that maximum release rates for radionuclides placed in repositories be established and adequate safety standards be defined, presumably by the Environmental Protection Agency and the Nuclear Regulatory Commission. (17, 28, 44, 95) Another commenter requested that this Statement outline EPA's role in waste management.
(202-HEW)

Response

The Environmental Protection Agency is currently formulating standards concerning the management and disposal of radioactive wastes. The standards are expected to include specifications concerning release rates to the accessible environment. The Nuclear Regulatory Commission regulations with which DOE must comply will reflect the Environmental Protection Agency standards. A discussion of applicable existing standards is presented in final Section 3.3.

LICENSING AND THE DECISION-MAKING PROCESSIssue

Several commenters pointed out that the Statement does not identify the "decision makers" in the waste management process. (63, 208-NRC)

Response

In the President's February 12, 1980 speech on radioactive waste management, he stated that for disposal of high-level radioactive waste he was adopting an interim planning strategy focused on the use of geologic repositories which would be able to accept waste from reprocessing and unprocessed commercial spent fuel. The interim strategy is required since full environmental review (according to NEPA) must precede final decisions on many steps which need to be taken. Following these environmental reviews, the President will reexamine the interim strategy and determine if any changes need to be made. Issuance of this environmental impact statement is intended to serve as a basis for that reexamination. See also response above on regulations.

Issue

Several commenters pointed out, that the Department of Energy should consult with the public and state and local governments in the selection of a proposed disposal method (and corresponding disposal site) as well as in solving of related waste management problems (i.e., transportation). (19, 22, 28, 37, 41, 43, 97, 98, 123, 129, 139, 156, 191, 204, 218-DOI)

Response

DOE agrees with this comment. During the process of preparing draft and final environmental impact statements on commercial waste management program alternatives, state and public comment and participation has been, and will continue to play a role in the decision-making process. Problems in related areas of concern (e.g., transportation) will be solved with input from both the public at large and state governments. In addition the President has recently created by executive order, a State Planning Council made up of Federal and state officials, chaired by a state governor (Governor Riley of South Carolina) whose function will be to advise the Executive Branch and to work with the Congress to address radioactive waste management issues, such as planning and siting, construction, and operation of facilities.

Issue

One commenter stated that the Statement does not identify how future national decisions will impact waste management. (22)

LICENSING AND THE DECISION-MAKING PROCESSResponse

The Statement analyzes several parameters which are intended to represent possible future situations resulting from national decisions. Five different growth scenarios are examined:

- Present inventory (equivalent to industry shutdown).
- Present capacity to retirement (equivalent to licensing no new reactors).
- Installed capacity of 250 GWe in year 2000 and declining to zero in year 2040.
- Installed capacity of 250 GWe in year 2000 and continuing at 250 GWe to year 2040.
- Installed capacity of 250 GWe in year 2000 and growing to 500 GWe in year 2040.

The Statement also examines two fuel cycle alternatives (see Section 3.2). The once-through fuel cycle is consistent with the present administration policy on reprocessing. The uranium-plutonium recycle option that is addressed could become viable if the moratorium on reprocessing were to be lifted.

Issue

Several commenters noted that the Statement does not discuss the process and schedule being used to resolve radioactive waste management questions. (43, 97, 147, 154, 198, 208-NRC)

Response

The Statement is intended to be the environmental input to the National Waste Management Plan which was called for by President Carter in his statement of February 12, 1980, and which will be issued for the public review in the near future. This plan will outline the programmatic structure for the management of radioactive wastes. The overall schedule and major milestones for implementation of the program will be identified in this plan. In his message of February 12, the President called for the Department of Energy to mount an aggressive research and development program over the next five years to support waste solidification and packaging, and repository design and construction including experimental test facilities. Selection of a site for the first full-scale repository should be accomplished by about 1985 and should be operational by the mid-1990's.

Issues

Several commenters questioned the use of an environmental impact statement at this stage of program development for waste disposal. (181, 217) A question was also raised as to the validity of using a generic approach in this particular EIS. (214)

Response

DOE has prepared this generic environmental impact statement in order to adequately examine the environmental impacts of alternative disposal strategies, including the

LICENSING AND THE DECISION-MAKING PROCESS

alternative of no action. This procedure is in accordance with accepted methods of complying with council on Environmental Quality (CEQ) regulations and the National Environmental Policy Act (NEPA). 40 CFR 1502.20 notes that agencies are encouraged to tier impact statements to eliminate repetitive material and focus on those issues that are pertinent at the given level of environmental review.

Issue

One commenter suggested that the Statement should note that radioactive waste disposal of any kind is prohibited or severely restricted in several states. (218-001)

Response

Since governmental jurisdiction of radioactive waste disposal is currently a matter of litigation DOE does not consider it appropriate to include in this Statement. The President in his February 12 statement on radioactive waste management stated "my administration is committed to providing an effective role for state and local governments in the development and implementation of our nuclear waste management program." By executive order, he subsequently established a state planning council to strengthen inter-governmental relationships to help fulfill joint responsibility to protect public health and safety in radioactive waste matters.

SITING ISSUES

Issue

Several commenters suggested the Statement note that during repository siting, an evaluation should be made of the effects on the states' historic, architectural, archeological, and cultural resources; on other potential energy sources (i.e., hydroelectric power); and on state and Federal laws. (4-FERC, 31, 39, 70, 184, 218-DOI)

Response

The NEPA process does provide the mechanism for ensuring that such concerns are addressed. Examination of potential sites will include a detailed examination of all the environmental impacts of the repository. This will include examination of the effects of the repository on historic, cultural, and natural resources and aesthetic values and on enforcement of applicable state and Federal laws. Section 2.3 also points out that such concerns will be addressed in detail as part of the site selection process.

Issues

Several commenters stated that if a repository is sited in a particular state, the Federal government should:

- Give states or communities impact funds. (28, 139)
- Consider mitigating the effects of increased monitoring, escorting, and emergency planning responsibilities of the states. (39, 184)

Response

As part of the process for siting a waste repository in a state, consideration will be given to the social and economic effects of that siting, including measures for mitigating undesirable direct and indirect effects. This consideration would be detailed in environmental analyses prepared for a specific site and is beyond the scope of a non-site-specific generic statement such as this. If specific funding to mitigate impacts in states or communities appears appropriate, DOE will need to request authority from the U.S. Congress to provide such aid. The President has directed the Secretary of Energy to provide financial and technical assistance to facilitate full participation of the state and local governments in review and licensing proceedings.

Issue

One commenter stated that sites selected for waste disposal must satisfy all the critical technological constraints, even if sites are located on Federally owned property. (43) Another commenter noted that it is unreasonable to limit site selection to those sites for which government has "an inferred availability of title." Eminent domain would seem appropriate. (113-EPA)

SITING ISSUES

Response

While the Statement acknowledges that siting may be affected by factors beyond those of a technical nature, Appendix B states that "satisfaction of appropriate technical criteria and siting considerations is essential at each stage" of the site-selection process. Licensing requirements being developed by the Nuclear Regulatory Commission and siting criteria under development by DOE will require any site under consideration for commercial nuclear waste repositories to be examined in detail and compared against relevant technical criteria for demonstration of technical adequacy.

DOE also recognizes that State acceptance of a site is an important component in the waste disposal program. DOE is now and will continue to be consistent with the "consultation and concurrence" approach recommended by the IRG (IRG 1979) and the President in his February 12, 1980 message on radioactive waste management. This approach allows for State and local interests to be represented in the repository site-selection process.

Issues

Several commenters stated concerns with specific siting problems, including:

- Repository siting should minimize transportation problems by locating near rail lines. (43, 157)
- Transportation should not occur in a particular area, near population centers or near food and water supplies. (87, 135, 209, 218-DOI)
- Repositories should not be sited in a particular state or area. (15, 52, 93, 94, 97, 99, 100, 112, 115, 139, 148, 162, 173, 174, 175, 179, 204, 206, 207, 209, 218-DOI)
- A particular site, type of site or regional siting was advocated. (2, 6, 28, 35, 71, 74, 115, 116, 136, 139, 180, 181)
- Regional siting should not be used. (35)

Response

The actual siting of waste disposal facilities will include consideration of minimizing waste transportation problems during the site-selection process. Site selection or exclusion criteria are being developed by DOE and its contractors and are currently under review by a broad spectrum of reviewers. As stated above, site selection will consider the pertinent environmental and legislative issues.

The Foreword notes that it is proposed that several alternative sites for geologic disposal should be qualified in different geologic environments and host rock types prior to the selection of a preferred site for repository construction. Four or five of these sites could be qualified on the basis of exploratory or geophysical techniques by 1985 leading to an operating repository by the mid-1990's.

SITING ISSUESIssue

One commenter suggested that the final Statement include sufficient detailed information that would outline those requirements of the Federal Land Policy and Management ACT (FLPMA) that pertain to withdrawal of public lands for a waste disposal site. This commenter also suggested that an analysis of the trade-offs between storage of nuclear wastes and existing uses of public lands be given attention in the final Statement. (218-DOI)

Response

Legislation such as the FLPMA will be considered as part of the site selection process. Such concerns would be evaluated in specific environmental impact statements following selection of several candidate sites. Except for the spoils piles the surface could reasonably be restored to preconstruction use(s). The spoils piles, even if on the order of 2 km square would not significantly impact on most land use. It is unlikely that a repository would be sited in an area where removal of small parcels of land from present use would constitute a substantial impact.

Issue

One commenter was confused by the repository site characterization process outlined in the draft Statement and raised a question as to the land use arrangements that will be required prior to test coring at a site. (181) Another commenter felt that more information should be provided on the site evaluation process. (218-DOI)

Response

A revised and more detailed discussion of the site exploration and characterization process appears in Section 2.3 of the final Statement. The DOE Position Paper to the NRC rulemaking proceedings (DOE 1980a) contains a thorough presentation of the Department's program for identification, characterization, acquisition and selection of potential repository sites.

Issue

One commenter noted that from the analysis presented, it appears that nonradiological environmental impact considerations will not influence the selection among the six geological disposal options for a given fuel cycle operation. However, even if this is true, consideration of environmental impacts will be important in site selection for any of the geological options selected. It is not readily evident whether one geological option should be selected before a comparison of alternative sites is made or whether, indeed, at a later date the site selection will include consideration of different geological options. Considerations of this type should be part of the "programmatic strategy" selection to be supported by the GEIS. (208-NRC)

SITING ISSUESResponse

In keeping with the IRG's recommendation and the President's message, a balanced approach is proposed with respect to repository media and site selection. As presently viewed, it is believed unlikely that a satisfactory repository in terms of media would not be acceptable on terms of site specific environmental impacts.

FOREIGN WASTES

Issues

Several commenters stated that foreign-generated commercial waste:

- Should not be transported to the U.S. (2, 98, 186)
- Should not be accepted if it cannot be properly disposed of. (62, 111)

Other commenters also asked who decides on shipment of foreign-generated commercial waste. (63, 186) Another commenter inquired as to whether such waste is presently being accepted into the U.S. (218-001)

Response

This Statement does not advocate a position on the receipt of foreign-generated waste. The decision to receive shipments of spent fuel from a foreign nation falls within the nation's international policy on nuclear fuels. Acceptance of foreign-spent fuels is anticipated to be based on one or both of the following criteria:

- The country is located in a region where protracted availability of spent fuel would be ill-advised in terms of nonproliferation objectives.
- Acceptance of the spent fuel would lead to significant nonproliferation gains (e.g., by encouraging alternatives to developing a national reprocessing capacity to meet spent fuel disposal needs, by stimulating implementation of desirable regional or international fuel cycle approaches consistent with overall U.S. policy, or by inducing adherence to the Treaty on the Nonproliferation of Nuclear Weapons) (IRG 1979).

Disposition of fuel which was supplied by the U.S. and used for power generation abroad is presently under the general direction of the Departments of State and Energy.

At this time, the U.S. is preparing to accept limited quantities of spent fuel from foreign nations.

Issue

Several commenters suggested discussing existing storage of foreign-generated wastes. (22, 167)

Response

The Statement does not address the subject of receipt and storage of foreign commercial wastes in detail. The estimation of the incremental foreign waste volumes is highly speculative; however, the characteristics of waste transported to the U.S. from abroad would not differ appreciably from domestic wastes. As a result, technical considerations presented in the Statement would apply to receipt and storage of foreign-generated wastes. The document DOE/EIS-0015 (DOE 1980b) discusses in more detail the impacts and issues associated with the receipt of foreign spent fuel.

FUEL CYCLE ISSUES

Issues

Several commenters stated that the Statement should include an analysis of:

- Head-end operations (mining and milling). (22, 39, 62, 155, 157, 184, 217)
- Low-level and defense or military waste. (22, 62, 97, 136, 198, 217)
- Breeder reactors. (39, 184)
- Reactor accidents. (62)
- Depleted uranium disposal. (196)

Response

The scope of the current Statement focuses on the impacts of the post-fission activities in the LWR fuel cycle. Defense wastes are addressed in the context of being additional wastes for disposal. Tables comparing key characteristics of defense wastes to commercial wastes is presented in Appendix I. Statements covering treatment and disposal of material (radiological and non-radiological) currently stored at three major DOE facilities have been (or are being) prepared for each site.

Other fuel cycle activities have been (or will be) addressed in separate DOE-prepared environmental impact statements (see final Chapter 2.0). Regulation of the management of commercially generated low-level waste (LLW) is not the responsibility of DOE. This function is performed by the NRC or by individual states having agreements to regulate LLW activities. (See Introduction for additional information.)

LONG-TERM STORAGEIssues

Several commenters stated that spent fuel storage:

- Should be limited to the extent possible to reactor sites. (45, 56, 78, 86, 122, 133)
- Could be at an Independent Spent Fuel Storage Facility whether offsite or at the reactor facility. (17)
- Either at reactor sites or at Independent Spent Fuel Storage Facilities should not be considered adequate waste management solutions. (35, 55, 157)
- In an above-ground facility is the best thing to do for the next 30 to 40 years. (158)

Response

Two EISs have been issued which discuss the impacts of spent fuel storage alternatives:

- DOE/EIS-0015, U.S. Spent Fuel Policy, May 1980 (DOE 1980b).
- NUREG-0404, Handling and Storage of Spent Light Water Power Reactor Fuel, Office of Nuclear Materials and Safeguards, March 1979 (NRC 1979a).

Presently, spent fuel is stored by the utilities at the reactor facilities. Under the President's Spent Fuel Storage Program, the U.S. Government would accept title, store retrievably, and dispose of commercially-generated spent fuel for a one-time charge. The use of an Away-from-Reactor Independent Spent Fuel Storage Facility (ISFSF) or an Away From Reactor (AFR) storage facility will permit programmatic flexibility in the development and scheduling of operational HLW disposal facilities. Storage at reactor sites and at Independent Spent Fuel Storage Facilities will not, however, be allowed to delay progress toward a licensed waste disposal repository.

DOE is in agreement with the third item in that an "adequate solution" implies permanency. As discussed in the Statement a no action alternative (leaving spent fuel at reactor sites or at AFR facilities indefinitely) is not considered a reasonable alternative.

ATTRIBUTION OF COSTS/RISKSIssue

Several commenters pointed out that the Statement should discuss the idea that the total impact of nuclear power (cost, risk) should be attributed to the project (beneficiaries) that created them and that the entity generating waste should be responsible for its disposal. (2, 3, 8, 9, 36, 55, 74, 77, 112, 116, 151, 155, 163, 193, 205) Other comments suggested nuclear waste disposal should not be financed by the Federal government since this acts as a subsidy of nuclear power. (42, 68, 163, 193)

Response

The Statement develops the waste management costs (1978 dollars) for the entire reference nuclear power generating system. Costs associated with treatment, interim storage, transportation, decommissioning, and disposal in geologic repositories are presented (see Sections 4.9, 5.6, and 7.6). The Statement identifies the dollar value a consumer of nuclear power could be charged for the waste management aspect of nuclear energy production. An assumption made when developing cost estimates was that all waste management costs, whether the services are provided by private industry or by the government, will be borne by the consumers of the electric energy generated by the nuclear power facility and thus are reflected as an increase in cost of power. The Department of Energy has, in addition, published estimates of proposed charges for spent fuel storage and disposal services in DOE/EIS-0015 (DOE 1980b). The stated basis for these estimates is full recovery of all waste management costs including research and development expenditures.

ALTERNATIVE DISPOSAL ISSUESIssues

Several commenters stated that certain of the alternative concepts should:

- Receive additional funding and study in order to further investigate feasibility of concept (or some aspect thereof). (9, 11, 23-DOC, 28, 35, 36, 63, 84, 96, 113-EPA, 120, 180, 194, 196, 212)
- Be eliminated as potential alternatives for such reasons as cost, safety, or technical feasibility. (9, 28, 30, 36, 43, 62, 86, 88, 96, 121, 122, 123, 140, 141, 144, 177, 194, 196, 213, 216)

Response

This Statement has been prepared to analyze the environmental impacts of implementing a national strategy for the disposal of commercially-generated radioactive waste including the possible effects of implementing a number of alternative concepts for waste disposal. To the extent possible, with information currently available, these alternative concepts have been considered in this Statement.

As a result of analysis in this Statement, and independent analysis by the Interagency Review Group on Nuclear Waste Management (IRG, 1979), the Department of Energy proposes to continue to place primary programmatic emphasis on geologic disposal with other disposal technologies (e.g., seabed and very deep hole disposal) receiving less program emphasis.

Issue

Several commenters felt that more exotic or additional disposal concepts should be included and analyzed. (6, 119, 149, 158, 164, 187, 192)

Response

The concepts analyzed in the Statement are viewed as a comprehensive set of disposal technologies. Any other disposal options that have so far been proposed are variations on one or more of the techniques addressed in the Statement.

GENERAL COMMENTS

In addition to the comments on policy aspects of commercial waste management, a number of comments were received which advocated points of interest or suggested items that were beyond the scope of this document. A number of these comments are discussed here, in general terms.

A number of letters were received that suggested slowing down or halting generation of electricity by nuclear power to stop the production of any more nuclear waste. (42, 57, 60, 65, 68, 90, 91, 96, 98, 102, 103, 105, 106, 109, 110, 111, 117, 128, 131, 134, 137, 138, 140, 142, 143, 145, 153, 155, 159, 160, 162, 165, 169, 171, 172, 174, 186, 194, 197, 199, 204, 206, 212, 216) Other commenters stated that generation of nuclear power should be halted until there is an accepted means of disposal of such waste. (59, 67, 69, 74, 78, 93, 118, 123, 125, 127, 135, 141, 146, 149, 177, 185, 195, 209, 213) As the Introduction to the Statement points out, the issue of waste production has been addressed only from the standpoint of the waste management impacts that would be associated with various future industry growth scenarios (including present inventory only).

Several letters contained statements about the future of fuel reprocessing. (27, 29, 35, 55, 82, 128, 147, 153, 158, 180, 181, 198, 205,) At present, reprocessing is indefinitely postponed under Presidential directive and this Statement has considered alternative approaches to treatment and disposal of commercial wastes including those that would require no reprocessing. Consideration of an alternative in no way endorses the alternative. DOE agrees that the Presidential deferral of reprocessing would have to be rescinded before any such reprocessing could take place. The waste disposal program should be applicable to all potential waste and waste forms. Since reprocessing is deferred but not banned, reprocessing wastes are potential waste forms and have been considered in this generic impact statement. An analysis of the pros and cons of reprocessing spent fuel and recycling uranium and plutonium can be found in NUREG-ES-0002 (NRC 1976a).

Commenters mentioned the undesirability of a breeder reactor program. (29, 96, 153) The breeder analysis is not covered in the Statement nor is this Statement the appropriate forum for a discussion of the appropriateness of instituting such a program. Such discussions have been covered in another impact statement (ERDA 1975).

Other methods for generating energy or encouraging energy development (e.g., solar installations, fusion reactors, and various tax incentives) as well as analysis of implementing these methods (e.g., comparison of expected exposures from nuclear sources to non-nuclear sources) were advocated. (7, 15, 20, 29, 63, 67, 72, 74, 76, 83, 91, 96, 102, 105, 108, 117, 138, 141, 142, 145, 171, 172, 173) This Statement evaluates only the alternatives for disposing of post-fission waste from commercial nuclear power production. Discussions of alternate forms of energy production or mechanisms to stimulate energy production and their impacts are outside the scope of this document.

WRITTEN PUBLIC COMMENTSTECHNICAL ISSUES AND RESPONSES

The technical issues (regarding the draft Statement) identified in the comment letters and the responses to these issues are presented in this section. Technical issues are those which specifically address the content, presentation and/or the analysis in the draft Statement. Such issues, if within the scope of the document, resulted in either modification or revision of the text. The responses to these issues identify how and to what degree the issue has been incorporated into the final Statement. Within each of the eighteen technical topic areas, issues are presented sequentially according to the page number referred to in the draft Statement. Issues not addressing specific pages or sections appear at the conclusion of each topic area.

ORGANIZATION AND PRESENTATION

Draft p. iv

Issue

One commenter objected to a perceived implied accusation (paragraph 4, sentence 3) that interim away from reactor spent fuel storage is lacking a firm technological base. (181)

Response

This comment is the result of a misinterpretation of the topic being discussed in the referenced paragraph. The ISF referred to in this paragraph is not an away from reactor storage facility or an interim spent fuel storage facility but an intermediate-scale facility for a geologic repository development.

Draft p. 1.1

Issue

Several commenters did not feel that the two primary conclusions stated on p. 1.1 were completely substantiated by the information in the text. (43, 97, 208-NRC, 217)

An additional commenter noted that it has not been established that geologic repositories are the best option--only that they may be an acceptable option for the first phase of disposal. Therefore, conclusion (2) does not follow from conclusion (1). (218-DOI)

Response

As noted below, DOE has taken several steps to improve the organization and content of the Statement. While the Department is of the opinion that the two primary conclusions

- The disposal of radioactive waste in geologic formations can likely be developed with minimal environmental consequences, and
- Therefore, the program emphasis should be on the establishment of mined repositories as the operative disposal technology.

were a logical result of the analysis in the draft, the final Statement more fully supports these conclusions. DOE is not required to develop the "best" action from an environmental perspective before proceeding, an environmentally acceptable one is sufficient.

The proposed action outlined in this Statement calls for primary emphasis on geologic disposal with secondary emphasis on other disposal technologies.

ORGANIZATION AND PRESENTATIONDraft p. 1.3Issue

In presenting the IRG's "key characteristics of a near-term interim strategic planning base for high-level waste disposal" parts omitted from the "key characteristics" seem significant and should be included. (208-NRC)

Response

In revising the Statement the "key characteristics" outlined by the IRG were deleted and replaced with the pertinent portion of the President's February 12, 1980 message to Congress announcing a comprehensive program for management of radioactive waste.

Draft pp. 1.3, 4.22, 4.38Issue

Clarification of the meaning of short-term and long-term as pre-closure and postclosure of the repository should be made when the terms are first used. The difference between short-term and near-term is not clear either. On p. 1.3, third and fourth paragraphs, the meaning of the "near-term" and "long-term" consequences are mentioned. The explanation for near-term in this paragraph is the same as that given for short-term on page 4.22. (208-NRC)

Response

Near-term and short-term are considered synonymous. However, the terms used in the final Statement are near-term and long-term where near-term is pre-repository closure and long-term is post-repository closure.

Draft p. 1.12Issue

Several commenters stated that detail concerning ^{14}C was out of keeping with the rest of the introductory scope. (34, 181)

Response

The detailed example using ^{14}C was removed from the final Statement.

ORGANIZATION AND PRESENTATION

Issue

Several commenters provided numerous suggestions for improving the organization and readability of the Statement.

- Clearly indicate purpose and proposed action. (38, 154, 182, 198, 217)
- Organize volumes so that it is easy to identify major conclusions. (58, 124, 154, 182)
- Revise summary to include more information to support proceeding with geologic disposal. (13, 34, 154, 181, 182)
- Improve summary by highlighting differences between alternatives. (201)
- Write a summary volume describing reference system which is easily understood by layman. (219)
- Draft p. 3.1 - Do not give impression that rationale for proceeding with geologic disposal is the amount of effort (funding) previously applied towards the technology. (35, 58)
- Minimize likelihood that statements taken out of context will contribute to uncertainties regarding waste disposal. (58, 124)
- Identify adequacy of current technology to support immediate repository program. (38, 124, 167)
- Present information in a form that can be readily understood. (13, 58, 124, 167, 219)
- Come up with a condensed version that most people will be able to understand. (2)
- Make better use of summary tables after each major topic. (7)
- Decrease size of document. (201)
- Do not constantly use technical jargon which weakens the analysis, and expand the Glossary. (208-NRC, 218-DOI)
- Improve cross referencing to material in body of report as well as supporting volumes and edit to remove redundancy. (58, 124, 154, 208-NRC)

Response

In preparation of the final Statement, the Department of Energy (DOE) has taken several steps to be responsive to the recommendations on the organization and presentation of the draft Statement. First, the structure of Volume 1 was modified to focus on the proposed Federal action and to make more evident the extent of the analysis in the Statement.

ORGANIZATION AND PRESENTATION

This was accomplished by:

1. Outlining the purpose and need of the Statement (Chapter 2.0). This chapter discusses the intent of the document, the proposed Federal action, and the "decision territory" covered.
2. Identifying programmatic alternatives (Chapter 3.0) including a statement of a no-action alternative which the draft Statement did not do.
3. Developing Chapter 4.0 which discusses predisposal options and systems.
4. Emphasizing the proposed action (a program leading to a mined geologic repository) by discussing in a separate Chapter (5.0) with the presentation of disposal alternatives in Chapter 6.0.
5. Including a new Chapter 7.0 which discussed tradeoffs between the proposed action and the two alternative actions on a complete system basis.
6. Expanding the Glossary (Chapter 8.0) to include additional geologic, environmental, and waste technology related terms.

The Summary chapter was extensively revised so that the material would be more easily understood by the general public. In revising this chapter DOE was sensitive to the comments that the significant conclusions be highlighted in the Summary and that they be substantiated by the material in the text. An effort was also made to increase the overall clarity and readability of the document by reducing the page length, being consistent in the use of terminology, using summary tables whenever possible, and relegating supporting data or information to the Appendices.

The Statement is now consistent with DOE policy as directed by President Carter in his February 12, 1980, Waste Management Policy Statement and conforms to the DOE Position Paper for the NRC confidence rulemaking hearing activities (DOE 1980a).

The Statement has been edited to reduce the ambiguities and redundancies found in the draft. The method for referencing supporting documents was not changed. The present method is believed to allow adequate access to the material referenced.

The tables that follow identify the correspondence between sections of the final Statement and the draft. These tables will be of assistance to the reader during the remainder of Volume 3 when reference is made to either the draft or final Statements.

Draft pp. 3.1.173-214

Issue

Discussion of environmental impacts of predisposal activities (storage, packaging and transport) is confusing. Some variations with various deferrals of decisions are included. All together 14 different options and sub-options are included. This sort of proliferation simply complicates the presentation. (154)

ORGANIZATION AND PRESENTATIONVolume 1

<u>Section of Final Statement</u>	<u>Corresponding Section of Draft</u>
Foreword	Foreword
1.0	1.0
2.0	Foreword, 2.1
3.1	--
3.2	2.1
3.3	2.2,2.3
3.4	Portions of 3.1.3 dealing with risk and risk perspectives
3.5	Portions of 3.1.2 and 3.1.3 dealing with non-technical issues
4.1	--
4.2	2.1
4.3	3.1.4, 3.2, portions of 3.9 and Appendix L
4.4	3.1.4 and portions of Appendix M
4.5	3.1.4 and portions of Appendix N
4.6	3.1.4 and portions of Appendix O
4.7	3.1.5.3
4.8	3.1.5.3
4.9	3.1.5.3
4.10	3.1.5.3
5.1	3.1.1 plus portions of 3.1.3 relating to the multiple barrier concept
5.2	3.1.2, 3.1.3 and 3.1.6
5.3	3.1.5.1
5.4	3.1.5.1
5.5	3.1.5.2
5.6	3.1.5.1
5.7	3.1.5.1
6.1	3.3-3.10
6.2	4.0
7.0	Portions of 3.1.5.4

Volume 2

<u>Section of Final Statement</u>	<u>Corresponding Section of Draft</u>
Appendix A	Appendix A
Appendix B	---
Appendix C	Appendix C
Appendix D	Appendix D
Appendix E	Appendix E
Appendix F	Appendix F
Appendix G	Appendix G
Appendix H	Appendix H
Appendix I	---
Appendix K	Appendix K and Appendix Q
Appendix L	---
Appendix M	---
Appendix N	---
Appendix P	Appendix P

ORGANIZATION AND PRESENTATIONResponse

The presentation was simplified and shortened to include only the reference or example case for each activity.

Draft p. 3.1.214Issue

Impacts are integrated for waste treatment, transport storage and final disposal. Tables 3.1.84-87 should be checked to eliminate errors in the supporting tables. (154)

Response

Draft Tables 3.1.84 through 3.1.87 were thoroughly checked. Some values that may appear to be in error were the result of rounding off the value.

Draft p. 3.1.233Issue

The discussion referring to Tables 3.1.84-87 should emphasize that environmental impacts (radiological and non-radiological) associated with all of the fuel cycles and media considered are trivial. This perspective should be brought into the final EIS. (154)

Response

This perspective has been brought into the final Statement (see Chapter 7.0).

Draft Section 3.1Issue

Summary tables that show the range of environmental impacts for geologic disposal would be helpful. (202-HEW)

Response

Chapter 7.0 and the Summary present such tables.

Issue

Several commenters noted that the analysis in the Statement should emphasize the systems aspect of waste management. (201, 219)

ORGANIZATION AND PRESENTATION

Response

The draft Statement did analyze the entire cycle of post-fission waste management activities. However, to ensure that the reader is able to recognize the scope of the analysis carried out, the final Statement is structured such that the predisposal activities (waste treatment and packaging, waste storage, waste transportation, and decommissioning) are first presented (see Chapter 4.0), the disposal activities are then outlined in subsequent chapters (Chapters 5.0 and 6.0) with the system impacts presented in Chapter 7.

Issue

Several commenters stated that the title should show that the document deals primarily with the disposal of high-level wastes. (113-EPA, 198, 201, 218-DOI) It was requested the term "generic" be in the title. (181)

Response

The title does include the word "generic" in it. The chapter presenting the document's purpose and need (Chapter 2.0, Introduction) discusses the "decision-territory" of the Statement and the bounds of the analysis. This chapter also contains a discussion which notes that the analysis is generic in nature.

Issue

Several commenters noted that the overwhelming emphasis of the document on geologic disposal does not enable a legitimate comparison to be made. (113-EPA, 214, 218-DOI) Other commenters took exception to what they stated as "bias" of DOE toward a single method of disposal. (167, 217)

Response

The regulations for implementing NEPA require that an environmental impact statement be a full disclosure document and present whatever relevant information is available on each of the alternatives. To this end, each of the disposal options in the Statement is discussed to the extent information is available. In preparing the final Statement an effort was made to increase the depth of analysis (quantitative and qualitative) in the sections discussing alternative disposal technologies. The disparity in the number of pages addressing geologic disposal versus the alternative disposal options results from the significant difference in the existing data base between mined geologic disposal and other techniques.

Issue

The suggestion was made that in the concluding chapter, a comparison of alternative disposal schemes and the alternative systems should be prescribed. (201)

ORGANIZATION AND PRESENTATIONResponse

Section 6.2 and Chapter 7.0 of the final Statement present a system comparison of the disposal alternatives.

Issue

One commenter suggested that when deciding which course of action to follow, DOE should consider the CEQ regulations (40 CFR 1502.14) which require the identification of any preferred alternatives in the draft Statement. (208-NRC)

Response

The proposed action (a program leading to mined geologic disposal) and the preferred alternative are the same.

SCOPEDraft pp. iv and 1.2Issue

The Statement should provide a comparative analysis of regional versus national repository approaches as discussed in the IRG report. (43)

Response

A discussion of tradeoffs involved in siting regional repositories is presented in final Section 5.3.

Issue

Several letters commented on the relationship between the Statement and the IRG Report.

Draft p. 1.3 - The relationship of the conclusions of the IRG report to the final EIS should be made more clear. (154)

The GEIS should examine each of the national strategies discussed in the IRG Report in an explicit manner which permits an environmental comparison of the alternatives. (208-NRC)

Response

The relationship of conclusions in the final Statement to the IRG report has been expanded upon (see Section 3.1, Chapter 7.0, and Summary).

The proposed action (see below) encompasses Strategies I, II, and III as described in the IRG Report. The alternative action is consistent with IRG Strategy IV.

Draft pp. 1.10-11Issue

Several letters commented on the need to address the excess quantities of plutonium available in the year 2040.

All of the waste which enters into consideration in the EIS should be addressed. (26)

It is not immediately obvious why there is 1300 MT excess. (34)

Some qualitative statement about the plutonium on hand in the year 2040 probably ought to be made. (154)

On p. 1.10 (and elsewhere) the statement is made that a separate and distinct nuclear fuel cycle might be in existence to receive 1300 metric tons of plutonium by the year 2040. This "Alternative" fuel cycle would also produce radioactive waste. Although this disposition may appear to be possible, the more prudent approach would be to consider this excess plutonium to be TRU waste requiring safe disposal in a repository. However, if credit is to

SCOPE

be taken for use of the plutonium in this "alternative" fuel cycle, the disposal of radioactive wastes from this fuel cycle should be discussed. (208-NRC)

Response

In developing a basis for this generic Statement, DOE recognized that it is not possible nor even reasonable to attempt to identify the impacts of commercial nuclear waste management over the entire indefinite future. No one can predict whether or not nuclear power will be utilized for another 10, 50 or 500 years. DOE attempted to place a reasonable boundary on the system to be analyzed that would be sufficient in scope to provide the information needed to reach well-balanced decisions necessary for the waste management requirements that can be foreseen today. DOE assumed that the future course of actions can and will be reexamined periodically in the future. For example, if the burden of nuclear waste management should become so onerous that nuclear power generation should be discontinued, which is not a conclusion of the present analyses, the decisions and necessary actions can be initiated when and if that becomes apparent.

The intent of the analysis in the draft Statement was to account for the waste management requirements for the maximum projected nuclear power capacity reached by the year 2000 and the implied waste management commitment of allowing these plants to complete a normal life cycle and be decommissioned after 40 years of operation. It was assumed that other plants were likely to be added after the year 2000 but the waste management requirements for these plants were considered to be outside the boundaries of the system analyzed. Thus, it was reasonable to assume that plutonium recovered in excess of requirements for the system analyzed could be utilized by these other plants. But the waste management requirements for these operations were clearly outside the scope of the analysis. DOE did not speculate as to what these additional plants might be--light water reactors, fast breeder reactors, or some other type of nuclear facility.

In the final Statement, DOE considers a similar situation but has reduced the maximum installed capacity in the year 2000 from 400 GWe to 250 GWe as that now appears to be the maximum reasonably achievable. In addition, in the final Statement, DOE has included consideration of much lower growth possibilities, a case that analyzes the steady-state operation of a 250 GWe system through the year 2040 and a case that considers continued growth to 500 GWe by the year 2040.

An expanded discussion of plutonium utilization can be found in final Section 7.3.7.

Draft p. 1.15 and Section 3.1

Issue

Several commenters stated that sufficient reference was not made to the efforts of other nations in the area of waste management. (144, 147, 154)

SCOPEResponse

Such activities have been included where appropriate by reference (see Sections 5.1 and 5.2).

Issue

Several commenters noted that storage and transportation were not addressed sufficiently in the draft.

The few paragraphs which are devoted to transportation impacts may not be sufficient to analyze this issue. (7)

The hazards of storage and transportation of radioactive wastes should be addressed. (14, 145)

Determination of a direct impact relationship between the transportation factors discussed in Appendix N and the disposal techniques discussed in the body of the document is difficult. (43)

The standard environmental impacts associated with any transportation, such as air pollution, noise, and water quality impacts are important to evaluate. (43, 97)

No reference is made to the factors involved in transporting wastes, such as, adequacy and availability of present system risk, safety, etc. (43, 97)

The Statement does not consider that humans will handle this material on the way to storage. (126)

A relative lack of emphasis is placed on the handling and transportation of these radioactive materials prior to emplacement. (197)

The impact of the transportation of nuclear wastes should be included in any broad analysis of the risks and impact associated with nuclear waste disposal. (215)

An issue which is inadequately addressed is the transport of radioactive materials from production site to storage site. (216)

Response

The draft Statement did analyze impacts of waste storage and transportation (see Section 3.1.5.3 and Appendices M and N) including:

- radiological releases
- non-radiological releases
- resource commitments
- costs.

DOE recognizes that the text may not have given the reader a full appreciation as to the depth of the analysis; therefore, DOE has taken steps in preparing the final document

SCOPE

to alleviate this. The restructured outline devotes a chapter to predisposal systems (Chapter 4) and in Sections 4.4 and 4.5 describes the waste storage and transportation systems. Sections 4.7, 4.8, 4.9, and 4.10 present the environmental impacts, accident analysis, cost analysis, and safeguard requirements for storage and transportation operations (as well as the other predisposal activities). Section 4.2 identifies the relationship between the various predisposal activities (including storage and transportation) and the disposal concepts that are analyzed in Chapters 5.0 and 6.0.

In addition, DOE has prepared a separate EIS which addresses the impacts of storage and transportation of spent fuel (DOE 1980b).

Issue

Several commenters stated that the Statement should address the implications of delaying a decision on the disposition of nuclear wastes (i.e. giving the matter further study or deferring the decision to select a disposal alternative). (147, 154, 217)

Response

The final document examines three basic waste management program alternatives. The proposed action is for DOE to maintain its primary research and development emphasis on mined geologic disposal with some secondary efforts directed towards other concepts. The alternative action would be then for DOE to adjust (or reorder) their R&D priorities. Presumably mined disposal would receive less emphasis and one or more of the other disposal concepts would be investigated more vigorously than under the proposed action. The third program alternative considered was no action, defined as a continuation of present action (water basin or AFR storage) with no steps taken by the Federal government to provide for final disposal of commercial wastes.

Implicit in the structure of the proposed action and the alternative action is that the point at which a geologic disposal facility would become operational would occur earlier under the proposed action. Chapter 7 examines the tradeoffs between the various program alternatives and, as a result, addresses the issue of a delay in the decision on the disposition of nuclear wastes.

Issue

Several letters noted that the subject of decommissioning should be addressed in the Statement.

Decommissioning should be discussed thoroughly in this document. (62)

No mention is made of cladding of nuclear power plants. No mention is made of shielding equipment necessary for handling processing, transportation and storage. (144)

Disposal of dismantled nuclear plants hasn't been resolved. (155)

SCOPE

The problems or costs associated with eventual decommissioning are not discussed.
(167)

Consideration should be given to ultimate disposal of activated reactor parts in the repository. (196)

Response

The draft Statement did examine the impacts of decommissioning retired facilities, including nuclear plants (see Appendix O). To ensure that the reader is aware of the systems nature of the analysis the final Statement presents the predisposal systems in a separate chapter (4) and the decommissioning activities in particular in Section 4.5.

Decommissioning activities related to disposal are discussed in the chapter on geologic disposal (Chapter 5) and in the section on alternative disposal concepts (Section 6.1) to the degree that information was available. The cost figures cited in Chapter 5 (see Section 5.6) and in Section 6.1 do include the estimate of the costs associated with eventual decommissioning.

Issue

Several commenters stated that an analysis of a no action alternative should be included in the Statement. (34, 38, 113-EPA, 154, 167)

Response

A no action alternative is discussed in the final Statement (see Section 3.1). This alternative is defined as continued storage of spent fuel elements in water basins (either using AFR storage or at reactor water basins).

Issue

One commenter recommended the environmental aspects of delayed commitment of wastes to the repository be discussed in the final Statement. (208-NRC)

Response

The proposed action (see p. 37) examines the impacts of repository availability for the years 1990-2010. The alternative action (see p. 37) examines the impacts of repository availability for the years 2010-2030.

RADIOLOGICAL ISSUESDraft p. 1.19Issue

One commenter mentioned that the Summary quotes a range of 50-500 somatic effects per million man-rem. The conservatism of the larger of these two numbers should be pointed out in view of the current BEIR Committee activities. (166)

Response

The values cited were arrived at after much deliberations over available data. The values reflect an estimate of the upper end of the range of possible health effects from small doses delivered over long periods of time at low dose rate. The values are believed to be conservative as suggested; the other end of the actual range may include zero effects.

Draft p. 1.19Issue

One commenter felt that the Statement assumes a rather small range of 50-300 incidents of "multi-factorial" genetic disease over all generations per million man-rem of exposure. (40)

Response

The values given represent what is believed to be acceptable to most of the scientific community as conservative numbers of incidents of genetic disease resulting from exposure to very low levels of ionizing radiation.

Draft pp. 2.1 and 2.2Issue

The NRC final EIS on spent fuel storage, NUREG-0575, (NRC 1979a) should be cited. (208-NRC)

Response

This EIS is cited in Section 4.4.1.1 of the final Statement.

Draft p. 2.1.7Issue

Regarding the transfer of plutonium oxide powder from the reprocessing plant to a mixed oxide fuel fabrication facility, this comment states, "DOE transportation regulations

RADIOLOGICAL ISSUES

presently require at (sic.) whenever plutonium is shipped, it must be shipped in a nitrate solution. Transport of oxide powders is expressly forbidden." (147)

Response

On the contrary, except for less than 20 Ci amounts, the only form allowed for plutonium shipment is as a solid (oxide or metal) (10 CFR 71.42).

Issue

Several letters requested that additional information be provided on background radiation levels.

Draft p. 2.2.1--Some baseline exposure needs to be presented so the reader will know what increase and what risks are entailed by additional exposures. (62)

Draft Appendix F-- Examples of existing radioactivity levels in the biosphere (surface, water, drinking water, air, etc.) would illustrate conditions prior to the operation of a waste disposal facility. (43)

Response

The discussion of natural background has been modified in order to give the reader a better feel for the existing environment in terms of the existing radioactivity levels (see final Section 3.3).

Draft p. 2.2.1

Issue

The document neglects to mention overall guidance provided by the FRC: Radiation Protection Guidance for Federal Agencies, 25 F.R. 4402 et seq. (5/18/60), for which 10 CFR 20 is one of several implementing regulations. (113-EPA)

Response

The cited reference was added in the final Statement (see Section 3.3).

Draft p. 2.2.2

Issue

One commenter noted that 10 CFR 20 only gives limiting concentrations. That means that a utility may put out as much radioactivity as it wants as long as it dilutes it sufficiently. This merely spreads the cancers throughout an entire population. It does not place a limit on how many cancers nuclear power may produce. (30)

RADIOLOGICAL ISSUESResponse

The concentrations in 10CFR20 are limiting concentrations, however, the principle of As Low As Reasonably Achievable (ALARA) as given for reactors in 10CFR50, Appendix I, and in 40CFR190 for fuel cycle facilities, also apply and guard against such population exposures.

Draft p. 2.2.2Issue

The formula given for allowable whole-body radiation dose should be checked for possible error. It is given as $5(N-18)$, N being age. The meaning is unclear. (218-DOI)

Response

This formula is correct as given. It represents allowable life time dose limit.

Draft p. 2.2.3Issue

No mention is made of EPA's regulations developed under the regulatory authority of the Marine Protection, Research and Sanctuaries Act of 1972 (Public Law 92-532). This authority should be referenced in this section. (113-EPA)

Response

The cited reference was added in the final Statement (see Section 3.3).

Draft p. 2.2.3Issue

Under EPA Uranium Fuel Cycle Standards, the last sentence in this section is in error. The effective date for application of 40CFR190 can be found in 40CFR190.12. This error should be corrected. (113-EPA)

Response

The sentence was revised.

RADIOLOGICAL ISSUESDraft p. 2.2.3Issue

The way in which the EPA drinking water regulations would be applied, if at all, is not made clear. These regulations are not directly appropriate to the disposal of radioactive waste since they do not control the contamination of the environment. They are directed toward a water supplier and applied to monitoring and corrective treatment regardless of the source of the contamination. The draft Environmental Impact Statement relates to activities of persons whose contamination of the environment is being limited. (113-EPA)

Response

Reference to EPA drinking water regulations was removed.

Draft p. 2.2.4, line 6Issue

This should be corrected to read: (b) Gross alpha particles activity (including ²²⁶Ra but excluding radon and uranium)--15 pCi/l. (113-EPA)

Response

The commenter is correct. See response above.

Draft p. 2.2.5Issue

Under "Clean Air Act Amendments of 1977" the text states: "The administrative and legal problems arising from the potential conflict with NRC regulatory authority and procedures originating in the Atomic Energy Act of 1954 have not been resolved. However, it is unlikely that existing EPA radiation standards will be changed, although administrative requirements may." This statement is presumptuous and does not reflect the major effort underway at EPA to develop regulations under the Clean Air Act, as amended. The text should be revised in the final EIS. (113-EPA)

Response

The cited sentences were removed from the final Statement.

RADIOLOGICAL ISSUESDraft Section 2.2Issue

Several commenters noted that in preparing the final EIS, reference is needed to the present development by EPA and NRC of Federal guides for radioactive waste management and standards for high-level radioactive waste. (113-EPA, 154, 198, 202-HEW)

Response

Mention of their development appears in Section 3.3 of the final Statement.

Draft pp. 2.3.2 and 2.3.3, Tables 2.3.1 and 2.3.2Issue

Both of these tables are taken from an obsolescent reference (ORP/CDS 72-1). More appropriate reference would be EPA report ORP/SID 72-1 (Reference 21, draft Section 2.3) with the cosmic ray doses augmented by the new information in NCRP Report #45 (Reference 10, draft Section 2.3). (113-EPA)

Response

Data from NCRP-45 (NCRP 1975) were used in the final Statement. (ORP/CDS 72-1 was used by EPA in EPA 520/1-77-009 released in September 1977).

Draft pp. 2.3.2 and 2.3.3Issue

One commenter noted that the tables on these two pages list the estimated cosmic-ray and natural terrestrial radioactivity doses to people in each state. The doses are dependent on latitude, altitude, and the presence of certain elements in the earth's crust. Measurements were obtained at many locations, and the results averaged within each state, then reported as an average value for the state. Since there are often large differences within a state, the results should have been reported in a way to display this, as by a contour map. A portion of a northern state that contains rock-bearing radioactive material, and is at a high elevation may not be acceptable as a disposal site. In California, there could be large differences between the dose received by a person at Lake Tahoe and one in the Imperial Valley. (214)

Response

Natural background radiation dose was presented only as a perspective against which other doses could be compared.

RADIOLOGICAL ISSUESDraft p. 2.3.3Issue

According to the note at the bottom of Table 2.3.2 some 39% of the political units are assumed to have the national average annual dose. This assumption is unwarranted. (147)

Response

The material in the distribution of dose from natural background appearing on p. 2.3.3 has been extensively revised. The refinement as dose distribution by political unit was no longer considered relevant and was removed.

Draft p. 2.3.4Issue

The dose estimates for radon are obsolete. Currently, dissolved radon in the body would give a dose of about 2 to 3 mrem/yr and the range of estimated dose from inhaled radon and daughters at 0.7 pCi/liter would be 130 mrem/yr to 1800 mrem/yr. See United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) Report (1977) for a more complete treatment of the question, also NCRP-45 (NCRP 1975). (113-EPA)

Response

The Statement notes that the whole body dose from dissolved ^{222}Rn is 3 mrem/yr. EPA states that dose is 2 to 3 mrem/yr. NCRP-45 on page 110 says dose to total lung is 90 mrem/yr and to segmental bronchioles is 450 mrem/yr. But Table 45 on page 109 gives 90 for ^{218}Po - ^{214}Po only plus 2 from ^{222}Rn and 3 for ^{210}Pb - ^{210}Po for a total of 95. (The range of 100 to 900 mrem/yr to lung would correspond to the average ^{222}Rn concentrations in some places in the U.S.) The value of 0.7 pCi/ (700 pCi/m³) was not found in either UNSCEAR report or NCRP-45. The values given in UNSCEAR, 1977, Table 30, page 80, are 160 (20-2000) for bronchial epithelial and 30 (3-300) for lung in rad/yr. If Q = 10, then these are 1600 mrem/yr for bronchial epithelial and 300 mrem/yr for lung.

Draft p. 2.3.5, Table 2.3.3Issue

The data in this table are obsolete, see UNSCEAR 1977 or NCRP-45 for current data. (113-EPA)

Response

Data from NCRP-45 (NCRP 1975) were used in the final Statement.

RADIOLOGICAL ISSUESDraft p. 2.3.5, Table 2.3.4Issue

This table is obsolete. Use natural background as summarized in NCRP-45. (113-EPA)

Response

Data from NCRP-45 (NCRP 1975) were used in the final Statement.

Draft p. 2.3.5, Table 2.3.4Issue

One commenter noted that the 130 man rem/yr is at variance with the 102 man rem/yr in BEIR I. (8)

Response

In final Section 3.3 natural background is given as 65 to 125 man rem/yr depending on location in the country. These values were obtained from NCRP-45 (NCRP 1975). The value in Table 2.3.4 was taken from an EPA document. However, EPA in commenting on the draft suggested use of NCRP-45 as more appropriate.

The discrepancy with BEIR I was not investigated.

Draft p. 2.3.5Issue

One commenter noted that in discussing radioactivity DOE states "Presently the linear theory is unproven at low levels. For this reason, health effects numbers at low levels must be stated as a possibility, with another possibility being zero."

DOE does not mention the treatise of Dr. Karl Z. Morgan, noted health physicist, entitled "The Linear Dose is Non-Conservative" which suggests that low doses rather than having linear health effects result in a higher number of health effects than had previously been considered and higher than linear anticipates according to the BEIR report. The studies of Dr. Morgan should be included in DOE's section on radioactivity, especially when the theories at the opposite end of the spectrum are given recognition. (See p. 2.38 re N. A. Frigerio and R. S. Stone).

Also Dr. Ernest P. Radford, chairman of the BEIR Committee, testified before Congress (Feb. 1978) that his Committee had erred in 1972 in setting radiation standards and that in certain cases, the permissible limits were set at least 10 times too high. He said: "The human data obtained from populations exposed to highly ionizing radiation such as alpha

RADIOLOGICAL ISSUES

particles (high LET radiation) indicates that cancer induction at low doses is probably greater per unit of physical dose than at high doses..." (55)

Response

This controversy is still going on. The reader is referred to Appendix E for a more detailed discussion.

Draft p. 2.3.6

Issue

The use of the Congressional Research Service (Reference 23, draft Section 2.3) estimate of 200,000 defective children per year does not agree with current estimates of 9.5% to 10.5% incidence of genetic disorders in newborn (see UNSCEAR 1977, p. 519). The UNSCEAR estimates suggest that this estimate of 200,000 is at least a low by a factor of two.

Estimates of malignancies occurring each year are better obtained from the American Cancer Society annual publication "Cancer Facts and Figures--19xx." For example estimates have been: 395,000 deaths, 765,000 cases of cancer--1979; 390,000 deaths, 700,000 cases--1978; 385,000 deaths, 690,000 cases--1977; etc. (113-EPA)

Response

The basis for the estimate of persons born "with some type of physical or mental defect", as quoted from the Congressional Research Service, was somewhat different from the basis employed by UNSCEAR in arriving at an incidence of "genetic disorders", and the estimates are consequently not the same. A factor of two does not seem important, however, for the purposes of this very general discussion of the implications of natural radioactivity. Similarly, more current data on malignancies could be given but the general implications would not differ. The number 340,000 in the Statement should have been identified as deaths due caused by malignancies.

Draft p. 2.3.6

Issue

The use of Frigerio and Stowe as a reference should be put in context. Aside from using the same obsolete reference of natural background used in the DEIS which inflates the probable difference in background between areas of the country, the others neglect to consider the potential effect of other carcinogens in the work place and the environment. Some of these problems are highlighted in multiauthor sections on "Demographic Leads to High-Risk Groups" and "Environmental Factors" in the volume Persons at High-Risk of Cancer

RADIOLOGICAL ISSUES

(J. F. Franmeni, Jr., editor, Academic Press, New York, 1975). Little support is given for the assertions in the referenced paper.

In a more complete report by the same author (N. A. Frigerio, K. F. Eckerman, and R. S. Stowe, "Carcinogenic Hazard from Low-Level, Low-Rate Radiation," ANL/ES-26, 1973) where all methods and assumptions are given, there are several flaws. A major flaw is the assumption that "all forms of cancer show very similar doubling doses and closely similar increases in mortality rate per rad". This assumption is made contrary to the evidence in ICRP, UNSCEAR, BEIR, and other reports that variations in the susceptibility of tissue to induction of different forms of cancer by irradiation are quite large and not necessarily related to the marked variations in natural incidence of the diverse types of cancer.

There are also problems in the statistical analysis in ANL/ES-26: misuse or misinterpretation of the t-statistic, failure to use Scheffes' test or calculations of variance ratio to check the significance of the series of t-tests, and use of gross averages in the analysis.

In reality, the paper can be shown to be erroneous by inspection of Frigerio et al.'s source of cancer mortality data, NCI Monograph 33, Patterns in Cancer Mortality in the United States: 1950-1967. In Monograph 33, Burbank presented an analysis of Dynamic Geographic Distribution for each cancer. The complex pattern of increasing and decreasing cancer mortality by state and cancer show that factors other than background are the major driving force in cancer mortality rates and that natural background radiation is not.

Indeed, in a later publication (A. P. Jacobson, P. A. Plato and N. A. Frigerio, "The Role of Natural Radiations in Human Leukemogenesis," Am. J. Public Health, 66, p. 31-37, 1976), a more reasonable major conclusion was reached: "It appears that conditions relative to populations and their environment could mask a radiation effect, if in fact one is present." (113-EPA)

Response

References 25 and 26 were deleted.

Draft Section 2.3.3

Issue

Several commenters suggested that a discussion of the observed health-effects caused by differences in natural radioactivity between different regions or states should be included in the Statement. (10, 198)

Response

The general public is probably too mobile to permit development of the epidemiological data in quantity and certainty in order to identify effects at these low levels. However,

RADIOLOGICAL ISSUES

it has been suggested in the recent DOE Position Paper to the NRC rulemaking hearings on nuclear waste storage and disposal (DOE 1980a) that normal variations experienced in natural background could reasonably serve as an upper bound for allowable standards for the impact of waste management.

Draft p. 3.1.35, First ParagraphIssue

This paragraph suggests that the fission product problem terminates with the decay of ^{90}Sr and ^{137}Cs . While this may be substantially true for the production of heat and for acute radiation hazard, it is not true for the significance of the waste as a health hazard. Doses from ^{99}Tc , ^{129}I , and ^{135}Cs are not negligible over a long time frame. This problem is repeated in the last sentence of the next paragraph. (113-EPA)

Response

The wording "majority of the dose": was not intended to imply what the commenter says it does. Doses from longer lived nuclides, ^{99}Tc , ^{129}I , and ^{135}Cs , are small because their abundance is low. Even at very long times the contribution to dose would be expected to be relatively small in comparison with that from ^{236}Ra .

Draft p. 3.1.38Issue

In the early phases, actinide elements, particularly ^{238}Pu and ^{241}Pu , are significant. Tritium (^3H) may also be significant. In the long time frame ^{135}Cs and ^{14}C might also be significant. (113-EPA)

Response

The paragraphs referred to are describing the division of radionuclides into two main groups; 1) short-lived fission products and 2) long-lived fission products and actinides. No attempt has been made to determine which actinides are significant with respect to time. The lists of nuclides presented in each paragraph is prefaced by the words "Some of the key isotopes". The significance of ^3H and ^{14}C is dependent on the fuel cycle and off-gas treatment system at reprocessing facilities.

Draft p. 3.1.38Issue

^{239}Pu should be considered a long-lived isotope. (17)

RADIOLOGICAL ISSUESResponse

A typographical error. ^{238}Pu should have been ^{239}Pu .

Draft p. 3.1.54, Item 2Issue

The chemically separated high-level waste to be considered must include the ^{129}I (and the other volatiles) recovered from the fuel reprocessing plant as required by EPA's Uranium Fuel Cycle regulations (40 CFR 190, 10(b)). (113-EPA)

Response

The wording in question has been changed to avoid this misunderstanding. Control of ^{129}I and other volatiles is indeed required. While ^{129}I is not included in the high-level waste, it is recovered in the reference waste management concept and sent to the repository with the TRU wastes. See Section 4.3 of the final Statement.

Draft p. 3.1.60Issue

The quote of 10 CFR 50, Appendix F, is in error. The regulatory policy stated therein is that liquid wastes at a reprocessing plant must be converted to a dry solid which is "...chemically, thermally, and radiolytically stable to the extent that the equilibrium pressure in the sealed container (required before shipping) will not exceed the safe operating pressure for that container during the period from canning through a minimum of 90 days after receipt (transfer of physical custody) at the Federal repository." (208-NRC)

Response

The quote of 10 CFR Part 50, Appendix F was corrected in the final Statement.

Draft p. 3.1.122Issue

One commenter suggested that the discussion of the interaction of run-off from storage piles as well as water pumped from the mine with freshwater systems should include a presentation of appropriate state and Federal discharge parameters. (43)

Response

State and Federal discharge limits could have been discussed in this generic Statement. However, the usefulness of including such limits is questioned.

RADIOLOGICAL ISSUESDraft p. 3.1.144Issue

Define "health effects" and assumptions for "translating" 1.8×10^8 man-rem into 1.8×10^4 to 1.4×10^5 health effects. (208-NRC)

Response

Health effects are defined in Appendix E, Volume 2 of the Statement; 100 to 800 health effects per million man-rem equates to 1.8×10^4 to 1.4×10^5 health effects for 1.8×10^8 man-rem.

Draft Appendix CIssue

This appendix is grossly unsatisfactory. It concentrates heavily on doses to individuals and does not appear to recognize that more recent standards, although they may be expressed in terms of dose to the maximum individual, have population dose as part of their basis. Among such regulations are:

1. Limitations on releases of effluents from power reactors (Appendix I to 10 CFR 50);
2. The uranium fuel cycle standards (40 CFR 190); and
3. The drinking water standard (40 CFR 141).

The limitations on releases of ^{85}Kr , ^{129}I , and transuranic elements, in 40 CFR 90, are explicitly based on population dose.

The general thrust of this appendix is that population dose is not a concept suitable for radiation standards. This is incorrect because the concept of ALARA usually involves balancing the cost against the reduction in population dose. It is perhaps significant that this appendix does not include any of the BEIR reports but limits itself to the 1969 report of the National Academy of Sciences. For currency, the appendix should consider additional references (e.g., References 1, 2, 10, and 16 from Appendix E) to bring the appendix up to 1977 at least. (113-EPA)

Response

DOE does not agree. The development presented in final Section 3.3.2 and Appendix C is based on individual dose which is a matter of historical record. No attempt was made to suggest that at very low doses and dose rates that population doses are not of major interest in exposure control.

RADIOLOGICAL ISSUESDraft p. C.2Issue

The paragraphs ending the section on "Background" and initiating "As Low As Reasonably Achievable Application" reflect some bias and a lack of candor in describing the use of risk coefficients in radiation protection. Almost all government agencies, particularly the EPA but including the NRC and the OSHA, have used or are using risk coefficients to estimate impact of radiation exposure. The ICRP (Reference 11, draft Appendix C) has gone entirely to a risk-based radiation protection system, using estimates of risk in optimizing radiation protection. ICRP has stated, "These risk factors are intended to be realistic estimates of the effects of irradiation at low annual dose-equivalents (up to the Commission's recommended dose-equivalent limits)" (ICRP Publication No. 23, 1978). The NCRP (Reference 15, draft Appendix C) seems to stand alone in its position discounting the use of linear, non-threshold risk coefficients in radiation protection. (113-EPA)

Response

Risk factors were used in estimating health hazards, but only to the extent that the incidence of cancer and genetic effects could be related to doses to population groups.

Draft p. C.3, Table C.1Issue

While the table is titled "Comparison Chart of Radiation Standards," it then lists "Standards or Criteria" and references ICRP and NCRP values or reports. ICRP and NCRP reports are recommendations or suggestions which may or may not be adopted or modified and adopted by national regulatory agencies. The references to ICRP and NCRP should be deleted from the table.

It should be noted, however, that there are ICRP reports pertinent to health effects. ICRP Publication 26 (Reference 11, draft Appendix C) and ICRP Publication 27 ("Problems Involved in Developing an Index of Harm," 1977), both provide recommendations on "acceptable" numerical risk estimates for radiation workers. (113-EPA)

Response

Because the work of NCRP and ICRP should be identified, the column heading has been changed to include recommendation; ICRP 26, 27 and 28 has been added under health effects.

RADIOLOGICAL ISSUESDraft Appendix CIssue

The discussion of the "as low as reasonably achievable" principle in this appendix is misleading in that it treats ALARA dose levels as fractions of maximum permissible dose levels for individuals. Instead, ALARA is primarily an analysis of risks to an entire affected population and of the cost-effectiveness of reducing that population risk. While ALARA individual dose limits can be derived for specific activities (e.g.; operating nuclear power plants), the most basic ALARA judgement concerns the cost-effectiveness of reductions in overall population risk (e.g., \$1,000 per man-rem, Appendix I). (208-NRC)

Response

The use of fractional dose levels is meant to be a practical approach to the ALARA philosophy. As inferred in the final paragraph, Appendix C.1 (and in agreement with the commenter's suggestion), use of the ALARA philosophy should be cognizent of dose reduction factors, cost effectiveness considerations, and should typically be evaluated on a case by case basis. The intent of the existing wording and use of ALARA does not appear to be in conflict with generally accepted definitions of ALARA.

Draft p. C.4Issue

One commenter noted that the Regulatory Guide "Calculational Models for Estimating Radiation Doses to Man From Airborne Radioactive Materials Resulting from Uranium Milling Operations" contains many errors and that these errors might be carried through into Appendix C. (30)

Response

Subject guide was not used in Appendix C.

Draft p. E.1Issue

The bias in selection of reference is obvious. While the last sentence quotes the NCRP and its dislike of linear nonthreshold risk and its use in radiation protection, to maintain balance the ICRP's use of risk factors as realistic estimates (see comment on Appendix C, p. C.2) for radiation protection and their use in ICRP Publications 26 and 37 should also be documented. EPA's policy statement, 41 F.R. 28409 (1976), should also be noted. (113-EPA)

RADIOLOGICAL ISSUESResponse

EPA contends that the quotation from NCRP in the final paragraph on draft p. E.1 reflects a bias, in that quotations more favorable to the linear nonthreshold approach to risk estimation could and should have been included. The NCRP quotation was, indeed, chosen because it represented the negative point of view, and it was the purpose of this particular paragraph to reflect that point of view. Most of the rest of the appendix consists of a positive application of the linear nonthreshold approach, with extensive documentation. It seemed only proper that the limitations of the approach be also cited.

Draft p. E.3Issue

In the discussion of BEIR risk estimates, emphasis is put properly on the range of uncertainty. However, mention is made of the BEIR Committee report that (Reference 1, draft Appendix E), "With this limitation in mind, the Committee considers the most likely value to be approximately 3,000 to 4,000 cancer deaths (or a 1% increase in the spontaneous rate)" (emphasis added). (113-EPA)

Response

EPA contends that, in the first full paragraph on draft p. E.3, reference should be made to the fact that the BEIR Committee, while acknowledging uncertainties, did state a "most likely value" of approximately 3000 to 4000 cancer deaths." The BEIR committee did not define what they meant by a "most likely value". For the proper appreciation of these risk factors, the unavoidable uncertainties should be stressed, not a false sense of certainty.

Draft p. E.3Issue

The paragraph considered only EPA's Uranium Fuel Cycle documents and states that the risk estimates there continue to be used by EPA. In reality EPA risk estimates have continued to change as new data becomes available. In addition to papers published by staff (e.g., Ellet, Nelson, and Mills, "Allowed Health Risk for Plutonium and Americium Standards as Compared with Standards for Penetrating Radiation," pp. 587-601 in Transuranium Nuclides in the Environment, IAEA, Vienna, 1976), various EPA reports (e.g., A Computer Code for Cohort Analysis of Increased Risks of Death, EPA 520/4-78-012, 1978, or Proposed Guidance on Dose Limits for Persons Exposed to Transuranium Elements in the General Environment, EPA 520/4-77-016, etc.) show updated risk estimates and how they were derived. (113-EPA)

RADIOLOGICAL ISSUESResponse

EPA contends that the risk factors attributed to the EPA in the second paragraph on draft p. E.3 have been superseded by more recently derived numbers, and a more extensive consideration of the various risk factors that have been suggested by the EPA, and by others, could have been included. In any case, the more recent EPA numbers would not have led to different conclusions with regard to the range of values employed in the Statement, as summarized in draft Table E.2 (draft p. E.6).

Draft pp. E.3 and E.5Issue

EPA's dissatisfaction with the health effects estimates in the Reactor Safety Study (RSS) is documented in Reference 53, draft Appendix E. Recent literature has done nothing to dispel our belief that the use of a dose rate reduction factor is ill-advised as is the minimal plateau duration (30 years) used in the RSS.

The UNSCEAR 1977 Report suggests (except for leukemia) a 50-yr expression period unless the period has been shown to be shorter or longer for a specific cancer (Reference 2, draft Appendix E, par. 12, page 363).

In particular, the two major human cancers associated with radiation, lung and breast, are considered to elevate risk for the duration of lifespan following exposure (C. E. Land, and J. D. Norman, "The Latent Periods of Radiogenic Cancers Occurring Among Japanese A-Bomb Survivors," Late Biological Effects of Ionizing Radiation, 1, IAEA, Vienna, pp. 29-47, 1978; V. E. Archer, E. P. Radford and O. Axelson, "Radon Daughter Cancer in Man: Factors in Exposure-Response Relationships," Health Physics Society Annual Meeting, June 1978).

The dose reduction factor in the RSS report appears to be derived from an analysis by Mays, et al. considering ten sets of animal data from nine studies. If an additional two studies (that happen to show a reverse effect) are included in the analysis, the dose reduction factor becomes 1.7 ± 0.20 reported by Mays, et al. As UNSCEAR 1977 points out, most of the existing animal carcinogenesis data comes from observations at doses above 50 rads and that each tumor-model system has peculiarities which prevent generalization across multiple organ systems and cancers. See Reference 53 of this appendix for comments on the dose reduction factor in the RSS.

As has been pointed out by Crump, et al. (K. S. Crump, D. G. Hoel, C. H. Langle, and R. Peto, "Fundamental Carcinogenic Processes and Their Implications for Low Dose Risk Assessment," Cancer Res., 36, pp. 2983-2979, 1976): "It is likely that the error in the acceptable dose associated with a simple linear extrapolation will be made less than those associated with species extrapolation to man from the laboratory animal data. The BEIR Report (Reference 16) recommended linear extrapolation on pragmatic grounds. The theoretical conclusions of the present paper are that linear extrapolation to low-dose levels is

RADIOLOGICAL ISSUES

generally valid as a realistic yet slightly conservative procedure" (emphasis added). That carcinogenesis by an external agent acts additively with any ongoing process is accepted by Crump, et al. and by Hilberg (Hilberg, A. W., "Low-Level Ionizing Radiation: A Perspective with Suggested Control Agency Options,") in 10th Annual National Conference on Radiation Control, HEW Publication (FDA) 79-8054, pp. 386-391, 1979) in his allusion: "And, conversely because man is living in an environment of chemical additives and pollutants, these may set the stage for action of a very small amount of radiation exposure."

Most of the arguments on the RSS report centered on low dose rate, low LET radiation. Alpha radiation dose response curves are usually characterized as both linear and dose rate independent (BEIR 1972; UNSCEAR 1977) or as possibly providing underestimated effects at low doses (J. Martin Brown, "Linearity versus Non-Linearity of Dose Response for Radiation Carcinogenesis," Health Physics, 31, pp. 231-245, 1976; V. E. Archer, E. P. Radford, and O. Axelson; "Radon Daughter Cancer in Man: Factors in Exposure-Response Relationships," Health Physics Society Annual meeting, June 1978). No reports except the RSS report considers a threshold curve a viable concept. (113-EPA)

Response

In this comment the EPA discusses its objections to a number of aspects of the Reactor Safety Study (WASH-1400), as reflected on pp. E.3 and E.5 of draft Appendix E. Since the Reactor Safety Study represents the conclusions of a respected body of scientists, many of whom were also members of the BEIR Committee, their conclusions deserve to be considered along with those of other experts. Appendix E does not adopt the values of WASH-1400 but merely considers them as part of the input from which the adopted values of Table E.2 are derived.

Draft p. E.4, Table E.1

Issue

A column in Table E.1, headed "Environmental Protection Agency," purports to be the risk estimates used by EPA. They are actually averages for various risk models used by EPA in reports and therefore are not directly comparable to the other risk estimates in the table.

The estimates of 54 leukemia deaths/10⁶ person-rem listed in the table were extracted from EPA 520/9-73-003-B (Reference 4, draft Appendix E). As stated in that report (p. A.14), the risk conversion factors are average values for absolute and relative risks in the BEIR Report, 1972. Moreover, they apply only to the dosimetric models used in EPA report 520/9-73-003-B.

RADIOLOGICAL ISSUES

The EPA risk for thyroid listed in Table E.1, 25 thyroid cancer death/10⁶ personrem, is referenced to EPA 520/4-76-017 (Reference 6, draft Appendix G). That risk estimate cannot be found in the cited reference. However, on p. 96, *ibid.*, it states "..., a population age weighted value of 60 thyroid cancers per million rems to the thyroid was used." A similar risk estimate is shown in Tables 45 and 46 of EPA report 520/9-73-003-C, Environmental Analysis of the Uranium Fuel Cycle. Part II--Nuclear Power Reactors, 1973. Note that these thyroid risk estimates refer to cases, not fatalities, and so do not fit into Table E.1 (113-EPA)

Response

In this comment the EPA notes certain qualifications on its risk factors as quoted in draft Table E.1 (p. E.4). The only substantive comment related to the value for the thyroid risk factor, which, as the EPA correctly observes, does not appear in the reference given. The number included in draft Table E.1 represented an attempt to translate the EPA "case estimate" as given in the quoted reference, to a "fatality estimate". In retrospect this seems ill-advised and the thyroid risk factor has been deleted, since EPA has not given such a factor in terms of fatal cancers.

Draft p. E.7 and p. E.8, Table E.3

Issue

Newcombe's estimate of ten genetic effects based on a normal incidence rate of 0.1% for autosomal/dominant disorders has not been supported by other studies. Current incidence estimates are about 1% autosomal dominant and X-linked disorders, the estimate in UNSCEAR 1977. (113-EPA)

Response

The genetic effects estimates attributed to the BEIR report and EPA in draft Table E.3 are not comparable to those given in the 1977 UNSCEAR report since they assume a 30-yr reproductive generation time. To compare the BEIR and EPA estimates with those of UNSCEAR, the BEIR and EPA estimates should be multiplied by a factor of about 0.6 to adjust for a 30-yr population generation versus the current, approximately, 50-yr population generation. More recent EPA estimates have been adjusted for the current population generation (Feldmann 1976), to yield 200 genetic effects, close to the UNSCEAR 1977 estimate.

EPA notes that some of the bases for Newcombe's estimate of a genetic risk factor, as included in draft Table E.3 (p. E.8), are not supported by other studies and have not been accepted by UNSCEAR. The fact that Newcombe's value "has not been generally accepted" is noted, as quoted, in the sentence on overlapping draft pp. E.7 and E.8. Newcombe's arguments still carry weight, however, and deserve to be noted as a responsible view, albeit one which is not employed in the Statement. The EPA also notes in this comment that the BEIR

RADIOLOGICAL ISSUES

and EPA genetic risk factors are not derived on the same basis as the UNSCEAR factors and that corrections can be made to eliminate this discrepancy. The same comment could be made for nearly every risk factor quoted in this appendix, but it seemed more appropriate to list the factors as originally reported. The corrections, in any case, would not alter the selection of values in draft Table E.2 and employed in the Statement.

Draft p. E.9, Table E.4

Issue

The column titled "Mays (19)" may contain an error. Mays, et al. (Reference 19, draft Appendix E) estimated 200 bone cancer deaths/ 10^6 person-rads. Using a Q of 10, this would be equivalent to 20 bone cancer deaths/ 10^6 person-rem. In a contemporary paper (C. W. Mays, Estimated Risk from ^{239}Pu to Human Bone. Liver and Lung, pp. 373-384 in Biological and Environmental Effects of Low-Level Radiation, Vol. II, IAEA, Vienna, 1976), Mays estimated (again assuming a Q of 10) 20 lung cancer deaths, 20 bone cancer deaths and 10 liver cancer deaths per 10^6 person-rem.

EPA in its guidance on transuranium elements (EPA 520/4-77-016) provided an analysis of the health impact of exposure to transuranium elements in the environment which includes both risk and dose-rate estimates for a cohort of 100,000 exposed since birth. This guidance is supplemented by technical reports, Technical Report EPA 520/4-78-010 and Technical Note CSD-78-1, which provide background information for the basic guidance document. Since the health impact calculated in these reports is based on lifetime exposure and risk coefficients for specific organs, the results are not indirectly comparable with Table E.4 but they are a more realistic estimate of health impact from transuranium elements in the environment. (113-EPA)

Response

The EPA suggests that the bone cancer risk factor of four bone cancer deaths/ 10^6 organ rem, attributed to Mays, in draft Table E.4 (p. E.9), may be in error. They note that the Mays reference gives a value of 200 bone cancer deaths/ 10^6 person-rads; and dividing by a quality factor of 10, the EPA suggests that the equivalent factor is 20 bone cancer deaths/ 10^6 person-rem. They neglect however, to apply the additional "distribution factor", n , which has a value of 5, and reduces the risk factor to four bone cancer deaths/ 10^6 person-rem, as listed in draft Table E.4. For surface-seeking alpha emitters in bone, 1 rad is equivalent to 50 rem, all doses being considered on an average-dose-to-bone basis. The later Mays paper referenced by the EPA gives the same estimate of 200 bone cancer deaths/ 10^6 person-rad.

The EPA also calls attention in this comment to its own risk factor for transuranics in bone, as developed for their Proposed Guidance on Dose Limits for Persons Exposed to

RADIOLOGICAL ISSUES

Transuranium Elements in the General Environment. Since this Guidance has appeared only for comment and has not been officially promulgated, it seemed inappropriate to cite it as an official EPA position. In any case, the risk factors proposed by EPA in that document differed only marginally from the Mays factors (e.g., 250 bone cancer deaths/ 10^6 personrad versus 200 bone cancer deaths/ 10^6 person-rad).

Draft p. E.10Issue

Although BEIR, 1972 did not provide a risk estimate for skin cancer, the 1978 Stockholm meeting of ICRP suggested if a skin cancer risk is required, an estimate of one fatal cancer per 10^6 person rem could be used. Averaging the risk estimates in UNSCEAR 1977, the skin cancer incidence is around 0.5 cases per year per 10^6 person-rem; with a 6% mortality this would be about two fatal skin cancers per 10^6 person-rem. The 1977 UNSCEAR Report suggests alpha risk might be higher. (113-EPA)

Response

The EPA correctly notes that, although the BEIR Committee considered numerical estimates of skin cancer risk unwarranted, as quoted on page E.10, data have been published from which such estimates could be derived. While UNSCEAR-77 discussed these data, it did not put forward a single risk factor. The ICRP at its 1978 Stockholm meeting did, as noted by the EPA, suggest a risk factor of one fatal skin cancer/ 10^6 person-rem, primarily because their weighted system of dose evaluation was logically incomplete without such a number. Although it would be possible to estimate fatal skin cancer incidence employing the ICRP risk factor, the EPA has not suggested that this be done. It seems more instructive to call attention to the general lack of significance of radiation-induced skin cancer than to produce numerical estimates of infinitesimal risk.

Draft Appendix EIssue

There is no mention of recent findings by the British National Radiological Protection Board or American National Academy of Sciences which were reported in the May 19, 1979 issue of the "London Economist." (104)

Response

Appendix E contains reference to the bulk of recent literature regarding the relationship between radiation exposure and possible latent health effects. It is believed that the material presented is in essential accord with the findings of the American National Academy

RADIOLOGICAL ISSUES

of Sciences. It may be noted that Appendix E provides ranges of health effects per unit quantity of radiation received by a population to reflect the range of views of authoritative bodies.

Use of findings from studies of the British National Radiological Protection Board becomes a part of the protection in America both through scientific exchange among individuals and via the International Committee on Radiological Protection.

Draft pp. F.15-17

Issue

Further significance of the pathway parameters used could be demonstrated if referenced to a model radioactivity surveillance program. (43)

Response

Even though null measurements will always be expected, a postclosure monitoring system will probably be established and observed for as long as future generations care to operate it.

The DOE Position Paper to the NRC rulemaking hearings on nuclear waste storage and disposal (DOE 1980a) identifies as a component of one of its performance objectives for waste disposal that active maintenance or surveillance for unreasonable times into the future not be a requirement.

Draft p. N.4

Issue

A reference to NRC/DOT/State surveillance program results would be useful for adding realistic perspective and credibility to the estimates of maximum driver and handler exposure in transportation. See "Summary Report of the State Surveillance Program on the Transportation of Radioactive Materials," NUREG-0393. (208-NRC)

Response

The reference suggested was not included in the final Statement. All the truck driver doses in DOE/ET-0029 and the Statement are based on the conservative WASH-1238 (AEC 1972) estimate.

Issue

The calculations of health effects in the draft understate the credible potential range by about a factor of two on the average because they use the BEIR I (1972) rather than the

RADIOLOGICAL ISSUES

BEIR III (1979) figures for cancer induction from low-dose, low-LET ionizing radiation which have been sex and age factored. (115)

Response

The 1979 draft of BEIR III was withdrawn. NAS released an updated BEIR III report in July of 1980 which indicates that risk estimates of cancer death from low levels of radiation are only half what they were thought to be in 1972 (and as reported in the BEIR I report). The range of conversion factors used in this Statement encompass the values suggested by both BEIR I (1972) and BEIR III (1980).

CONSEQUENCE ANALYSISDraft p. 1.6Issue

The statement, "... it is very probable that integrated nuclear waste systems can be designed to assure that current and future generations of man will not be subjected to undue levels of risk from radioactive wastes, has not yet been established in the report.

(218-DOI)

Response

DOE is of the opinion that this particular statement is supported by the body of the document. In preparing the final Statement, efforts were made to strengthen the ties between the material in the text and the conclusions outlined in the Summary.

Draft p. 1.16Issue

In the definition of risk, "magnitude of the loss" is better expressed as "consequences of the event." This will also make the definition of risk consistent with that used in footnote e to draft Table 1.4 and the footnote on draft page 1.21. (208-NRC)

Response

The DOE believes that there is not a sufficient difference between the two expressions to warrant a change.

Issue

Several letters commented on whether an assessment of the risk of radioactive waste disposal had been performed.

Draft pp. 1.16 and 1.20--We note that a risk assessment requires the identification of a broad spectrum of event probabilities and consequences. It is not limited to worst case consequence assessments as is indicated in draft Tables 1.3 and 1.4. (208-NRC)

The GEIS fails to perform a risk assessment; instead it performs a consequence analysis ("what if" calculation) resulting from four disruptive events. (217)

To multiply consequence times probability to yield a risk and then to say that the risks obtained represent the worst possible risks is not always proper. This is valid only where it has been demonstratively shown that a disruptive event with a lesser consequence but a higher probability does not have a higher risk (that being a product of two numbers) than the risks resulting from the high consequence but low probability disruptive events. The approach of the GEIS in obtaining risks is simply unsound. (217)

CONSEQUENCE ANALYSIS

Risk defined as the product of probability times consequences should be qualified to reflect the possibility of increased public concern where serious consequences are involved. (219)

Response

While risk assessments are not limited to worst cases, only cases believed to be "worst" were analyzed in the Statement. In its presentation, the Statement first outlines the radiological impacts of unintended events from a consequence viewpoint. The frequency of occurrence of the particular event is then identified so that if one desires to determine the expected impact from such an event one may do so. However, the consequences of a given event (in the absence of the frequency of occurrence) are certainly an upper bound estimate of the impacts.

Draft p. 1.17Issue

The difference between "major disasters" and "primary events" is unclear. (34, 208-NRC, 218-DOI)

Response

The breach of a repository by a major disaster has two sources of environmental damage. The first is due to the physical disruption of the landscape by the primary event; for example, crater formation in a meteorite impact or a nuclear weapons strike. A second source of damage is the radiological consequences of a release of radionuclides to the environment. For all disasters considered in this Statement except solution mining, the consequences of radionuclide releases are less than those of the initiating natural disaster.

Draft p. 1.17Issue

One commenter noted that the uncertainty of geologic prediction does limit the application of risk assessment. If the probability of a certain geologic event occurring is not known, how can a reliable risk assessment be calculated to include the potential impact of such an event. (218-DOI)

Response

DOE agrees. However, if the consequences remain insignificant over many orders of magnitude, then the uncertainty may not be important.

CONSEQUENCE ANALYSISDraft p. 1.18Issue

One commenter requested that the basis for the frequency of stream invasion of a repository should be presented. (218-DOI)

Response

The total probability of release of waste materials from a repository via stream invasion is 4×10^{-15} (for one year). This figure is the product of the following:

- A fault intersection in the repository- 4×10^{-11} /yr
- A failure of a waste container- 10^{-2} /yr
- Chance of aquifer intersection which leads to biosphere- 10^{-2} /yr

The value of 4×10^{-11} /yr is taken from H.D. Clairborne and F. Gera, 1974, Potential Containment Failure Mechanisms and Their Consequences at a Radioactive Waste Repository in Bedded Salt in New Mexico: Oak Ridge National Laboratory, ORNL-TM-4639.

A more conservative value of 2×10^{-13} /yr for the probability of release of waste via stream invasion (versus 4×10^{-15} /yr) was chosen for illustrative purposes in the draft. Because it is believed that such a value may not be defensible over millenia, a value of 4×10^{-11} /yr was used in this final Statement to assure conservatism.

Draft p. 1.19Issue

The chemical nature of waste and of the geosphere appears to be largely ignored. Much of the reduction of radiation dose appears to occur as a result of the delay of radionuclides by sorption from the groundwater. The sorption of radionuclides depends on several factors, including the oxidation-reduction state of the nuclide, the presence or absence of complexing or chelating agents, and the nature of the specific geological materials present. In some cases, particularly if large and rather exotic containment canisters are postulated, the ion exchange requirements of the canister materials may be quite significant and might overload the exchange capacity of the media in which the waste was emplaced. (113-EPA)

Response

In some accident scenarios ingestion and use of ground water would contribute to the maximum individual dose. In those cases these pathways were analyzed.

The chemical nature of the waste geosphere is not ignored. Retardation of radionuclides is dependent upon several factors. However, with respect to transport modeling of radionuclides, this Statement utilizes generally accepted modeling techniques. Of course

CONSEQUENCE ANALYSIS

Sr and Cs have been studied extensively, and as a result their chemical behavior can be modeled quite well in groundwater systems. The same is not true of the other nuclides.

Data is also a problem. Quite simply, available data is usually insufficient to perform this type of modeling with accuracy. What one is left with is the conservative approach. This was the approach taken for the Statement.

Draft p. 1.19Issue

It would be useful to provide any available risk (consequence x probability) estimates for the transportation accident being discussed. This will allow a comparison to be made with the risks for the other accident scenarios. (208-NRC)

Response

Accident frequencies are given in the accident description tables of Section 6, DOE/ET0028, Vol. 4, and in Section 4, DOE/ET-0029, Vol. 1. The basis for accident analysis is described in Section 3.2 of the final Statement.

Draft p. 1.19Issue

Several commenters did not agree with the Statement that "...the expected frequency (from a transportation accident) brings the risk to a negligible value." (62, 128)

Response

Risks from transportation of nuclear material, as discussed in this report, are low compared to other societal activities. See Sections 3.4 and 4.8 of the final Statement.

Draft p. 1.19Issue

The Statement says transportation risks are the same for all options. However, estimates of risk in the tables in the supporting documents do not seem to support this conclusion. (208-NRC)

Response

The text states that the differences in transportation risks are small for all options. A review of these risks indicates that the risks themselves are small. Therefore any difference would also be small or insignificant. See DOE/ET-0029, Section 4 for analysis of risks from transportation.

CONSEQUENCE ANALYSISDraft p. 1.19Issue

A credible event missing from the discussion is the possibility of a water well drilled into adjoining hydrostratigraphic units that could disrupt regional flowlines and equipotentials such that radionuclide migration may be increased. Leakage through overlying aquitards into more permeable units could significantly speed the movement of radionuclides to the biosphere. The pumping well in this scenario would not be pulling radionuclides directly into its cone of depression since most water wells are not at that depth nor would the repository be located in a productive aquifer of potable grade water.

Further, the discussion on solution mining and the missing scenario on deep drilling activities such as natural gas and oil exploration ignore the potential for groundwater hydraulic and pollution effects. (208-NRC)

Response

A case of well contamination has been added in Section 5.5.

Draft p. 1.19Issue

One commenter pointed out that health effects based on routine operations are not trivial to nonexistent. (62)

Response

Routine operations are defined as accident free situations. In this Statement 100 to 800 health effects are postulated to result for each million man-rem of whole body dose received. The collective dose from routine operations resulted in between 0 and 750 health effects (on a world wide basis for the 250 Gwe growth and decline system).

Draft p. 1.20 Table 1.4, Item 1Issue

Although the person closest to the repository will be killed, there will still exist a maximum individual who receives the largest dose as a result of the release. (208-NRC)

Response

DOE agrees. However, the maximum individual chosen is a worst case. Since this

CONSEQUENCE ANALYSIS

individual is killed by the meteorite strike, his dose was not included in the draft Statement. The dose to this person has been added in the final Statement.

Draft p. 1.20, Table 1.4Issue

(a) The potential for a dose due to airborne dispersion caused by a meteorite impact does not appear to have been considered, (b) the units of "Health Effects," e.g., acute fatalities, morbidities should be defined, (c) the units of "Risk," e.g., total health effects, health effects per year should be defined, and (d) a description of how "accident probabilities" were arrived at and an associated uncertainty should be presented, e.g., both the probability for meteorite impact and the probability for fault fracture and flooding were given as 3×10^{-13} . Including uncertainty in the estimates of probability is also important since point estimates of probabilities as low as 10^{-13} , are difficult to justify when little data is available. (208-NRC)

Response

For the meteorite-impact scenario, the airborne dispersion was considered (see DOE/ET-0029). Appendix E of the Statement provides a definition of health effects. Derivations of accident probabilities are contained in those sections of DOE/ET-0029 where the accidents are analyzed.

Draft p. 1.20Issue

One commenter noted that Table 1.4 summarizes the results of long-term radiological impact analyses presented later in Section 3.1.5.2 of the Statement. While ranges in the estimates for health effects are given, there appears little discussion of the uncertainties associated with the stated accident probabilities, particularly for meteorite impact and fault fracture and flooding. The corresponding uncertainties should be fully addressed in this presentation. (168)

Response

There is indeed little discussion on uncertainties associated with probabilities used. The information was not available. It is pointed out that errors in the probability estimate of several orders of magnitude would not suggest a significant mathematical expectation of societal risk (see final Section 5.5).

CONSEQUENCE ANALYSISDraft p. 1.20 and 3.1.169Issue

Several commenters questioned the accident probability used for the solution mining scenario. (30, 40, 58, 142)

Response

The cited probability was arbitrary and has been removed.

Draft p. 1.20Issue

Several letters commented on the probability assigned to the drilling event.

Draft p. 1.20--The use of 0.005 as the probability of drilling is not understood. (40)

Draft p. 3.1.165--The overall probability of a contaminated drilling event occurring can exceed 0.005, since multiple drilling events over a period of time are possible. (113-EPA, 114)

Draft p. 3.1.165-168--If drilling occurs, probability is 0.005 that interception of a canister occurs. Accident consequences are based on 1/4 canister brought to surface and distributed through top 5 cm (15 cm?) of 0.5 ha. Probability of drilling actually occurring in the area is not given. (154)

Response

The first sentence of paragraph 3, draft p. 3.1.165, states, "Because it is not possible to determine a probability for exploratory drilling it was not possible to assign an overall probability to this event." The probability of any one drill hole striking a waste canister is 0.005 based on areas involved. The text was changed to reflect this fact.

The probability of drilling in the repository is considered unlikely but has not been quantified.

Draft p. 1.20Issue

One commenter noted there is continued reference to "what if" and "design basis" accidents. The distinctions between these terms are never clearly drawn. (40)

CONSEQUENCE ANALYSIS

Response

The definition of design basis and non-design basis appears in final Section 3.2. The so-called "what-if" accidents are considered to be in the non-design basis category.

Issue

Several commenters questioned the time frames used in the accident analysis, recommending investigations of both earlier times as well as later times.

Draft p. 1.20--It is amusing to see the "summary of estimated long-term radiological impacts of worst case what-if accident scenarios for breach of repository 1000 years after closure" when no one knows how to calculate the likelihood of a nuclear war within, say, the next 20 years. (108)

Draft p. 1.20--It does not seem adequate to restrict the EIS to only 1000 years, when it is dealing with radioactive waste which will not decay for up to several hundred thousand years. (142)

Draft pp. 3.1.165, 168--Basing the drilling and solution mining scenarios on a 1000 year time interval is not defensible and causes significant alteration of the predicted dose consequences. These time intervals should be reduced, preferably to approximately 250 years. (198)

Draft pp. 3.1.165-172--These discussions do not represent "worst-case" scenarios because the intrusions take place 1000 years in the future rather than 500 or 100 years in the future. (114)

Response

The analysis of the impacts of long-term (non-design basis) accident scenarios at a geologic repository was generally performed at four points in time: at year 2,050, +1000 years, +100,000 years, and +1,000,000 years (solution mining was only examined at +1000 years). The occurrence of such events at the time of repository closure is highly unlikely. The use of +1,000 years was to contrast the impacts for shorter surveillance times. The maintenance of institutional control for 1000 versus 100 years was not contended. The later time frames provide additional points of reference with which to view the accident consequences. Examination of additional time periods would yield similar results.

Draft p. 1.20, Table 1.4 and p. 3.1.173, Table 3.1.54

Issue

There were a number of errors and inconsistencies in these two tables. (154)

CONSEQUENCE ANALYSISResponse

Tables 1.4 and 3.1.54 were removed.

Draft p. 1.36Issue

The conclusion that long-term radiological impacts are low for the general population was not supported by the text of the Statement. (213)

Response

DOE's position is that the information presented in the Statement relative to radiological impacts does support the conclusion that over the long term such impacts are anticipated to be low.

Draft p. 1.36Issue

One commenter questioned the statement, "and very few maximum individuals receive significant dose; thus long term risk is not a decisive factor." Does this statement imply that some persons are deemed to be "maximum" and therefore more important than others, or is it misplaced modifier? (213)

Response

The statement was poorly constructed. The use of "maximum" was a case of a misplaced modifier.

Draft pp. 3.1.28, 36, 41, 120Issue

The subject of occupational radiation exposure is not adequately addressed in the GEIS. It should be considered in connection with short-term environmental impacts and the probability of various accidents occurring during the handling and emplacement of canisters. (208-NRC)

Response

It is believed that additional information on occupational exposure from accidents could strengthen the Statement. Some view EISs as describing consequences to individuals located in the environs of a plant and not to plant workers. Some additional material has

CONSEQUENCE ANALYSIS

been provided in the final Statement. For example, the exposure that might result to the workforce from a canister drop down a mine shaft followed by a rupture of the canister was developed and presented in Section 5.4.

By and large, protection of the workforce is stressed in other documentation such as Safety Analysis Reports for specific plants.

Draft p. 3.1.41, last bullet

Issue

The use of adsorption coefficients from one set of Hanford subsoils, measured under laboratory conditions, is not an adequate basis for scoping the effect of adsorption. There are some substantial differences between the adsorption coefficients of the Hanford subsoil and of those given on page K-20 of the Waste Isolation Pilot Plant EIS (DOE/EIS-0026-D), for example. (113-EPA)

Response

At the time the work was done the Hanford, or typical western desert subsoil sorption data was the only set of data available for reference.

Draft p. 3.1.64

Issue

One commenter questioned the definition of risk as "the sum product of the magnitude of losses and the probability that the losses will occur." A tendency exists for aversion of high consequence accidents, which would imply a valuation other than a strict probability-consequence product. (113-EPA)

Response

The final Statement refers to the product of probability and consequences (or societal losses) or mathematical expectation of societal risk. This does not include "risk aversion" which is likely to vary from individual to individual. Where first introduced in the text the aspect "non-linearity of risk aversion" is addressed.

Draft p. 3.1.64

Issue

Consequence analysis for any release is the estimation of the effect of that release. It is not restricted to postulated worst cases. (113-EPA)

CONSEQUENCE ANALYSISResponse

Consequence analysis is indeed not restricted to postulated worst cases. The text was changed accordingly (see Section 3.4).

Draft p. 3.1.67Issue

Since the long-term behavior of the parameters is uncertain, risk assessment should be based on upper estimate predictions as well as on "reasonable" predictions. (113-EPA)

Response

It should be based on "reasonable" predictions plus an estimation of the uncertainties involved.

Draft p. 3.1.67Issue

Are the probabilities in the first paragraph best estimate probabilities, upper bound probabilities, or what? (113-EPA)

Response

The probabilities referred to are best estimate probabilities.

Issue

Several letters commented on the probability cited for the faulting and flooding accident.

Draft p. 3.1.67--Uncertainties and the method for determining them should be consistently included with probability and consequence estimates. Although uncertainties are discussed in isolated cases, they are usually not included with point values, e.g., the probability of faulting through the repository is estimated at 4×10^{-11} per year (pp. 3.1.67) with no indication of associated uncertainties. (208-NRC)

Draft p. 3.1.155--The value of 4×10^{-11} /yr for the probability of faulting or fracturing (H.D. Claiborne and F. Gera, 1974, Potential Containment Failure Mechanisms and Their Consequences at a Radioactive Waste Repository in Bedded Salt in New Mexico: Oak Ridge National Laboratory, ORNL-TM-4639) used in risk considerations is outdated and its uncritical acceptance is a major shortcoming of the draft. This is not to say that the values for faulting or fracturing probabilities ultimately used for a site-specific risk assessment will not be some low number such as this, but these probabilities will have to be determined

CONSEQUENCE ANALYSIS

on a sound basis. Research to do this should be identified in the section on research and development needs. (218-DOI)

Draft p. 3.1.155, last paragraph--This is an improper combination of probabilities. If the probabilities are multiplied together, as has been done here, the result is the probability of all three conditions occurring in the same year. If the probabilities are taken over 10,000 years, for example, the probability of a fault intersecting the repository is 4×10^{-7} . The probabilities of failure of waste containment, or of aquifer intersection, over this period are likely to be one, each. The total probability is therefore about 4×10^{-7} , not 4×10^{-15} . (113-EPA)

Response

The total probability of release, 4×10^{-15} is for one year. The probability discussed on draft p. 3.1.155 is referred to as "total probability" because it is the product of

- a fault intersection in the repository $4 \times 10^{-11}/\text{yr}$
- a failure of a waste container $10^{-2}/\text{yr}$
- chance of aquifer intersection which leads to biosphere $10^{-2}/\text{yr}$

The text was changed to make clear that the total annual probability is being referred to. In order to assure conservation, this final statement uses a value of 4×10^{-11} for the total probability of release for one year and a probability of 4×10^{-7} for the 10,000 year period, as suggested by EPA.

Draft p. 3.1.67Issue

One commenter noted that the statement pointing out that ^{90}Sr and ^{226}Ra dominate the health effects of a release of radioactivity from a repository needs discussion. (40)

Response

This statement is actually discussing consequence analysis. ^{90}Sr and ^{226}Ra were used as examples. The importance of these nuclides at different times can be found in final Sections 5.4 and 5.5 or in Chapters 4.0 and 9.0 of DOE/ET-0029.

Draft p. 3.1.67Issue

Several commenters questioned the probabilities in Table 3.1.3.

CONSEQUENCE ANALYSIS

These probabilities are for purely random events. They are completely in error for non-random events. (30)

The draft states that the Poisson process is used to model the occurrence of geologic events, based on past observation. However, whether this table presents the probability that one event occurs for the "interval" of concern or, more properly, that one or more event occurs during this period is not clear. From $P(x) = \frac{(gQ)^x}{x!} e^{-gQ}$, the probability of one or more events occurring is $(1 - \text{the probability of zero occurrences}) = (1 - P(0)) = 1 - e^{-gQ}$. This formulation, however, produces somewhat higher probabilities than those listed in draft Table 3.1.3, e.g., for the "number of occurrence years" equal to 10^6 years, and an "interval" equal to 10^4 years, the probability that one or more geologic event occurs is 9.95×10^{-3} as compared to 6.9×10^{-3} . Thus, more explanation of the probabilities in draft Table 3.1.3 is needed. (208-NRC)

Response

The probabilities listed in Table 3.1.3 are for random events. The discussion surrounding the use of this table relates to the possibility of geologic events that would disrupt the integrity of the waste repository. It is agreed that these probabilities would be in error for non-random events. However, it is felt that careful site selection to identify all potential containment failure mechanisms would preclude other than random geologic events with potential to cause a waste repository failure.

Draft p. 3.1.68

Issue

One commenter suggested referencing the model calibration done by J. B. Robertson (1974, Digital Modeling of Radioactive and Chemical Waste Transport in the Snake River Plain Aquifer at the National Reactor Testing Station, Idaho: U.S. Geological Survey Open-File Report (AEC-22054)). (218-DOI)

Response

More recent references were added in the final Statement, Section 3.4.3.2 (Cole 1979 and Robertson 1977)

Draft p. 3.1.69

Issue

The man-caused events listed should be discussed in depth. (114)

CONSEQUENCE ANALYSISResponse

DOE/ET-0029 presents a more detailed discussion of the release scenarios analyzed in the Statement.

Draft p. 3.1.70Issue

The last sentence in paragraph 3 contradicts the first two. (218-DOI)

Response

The subject paragraph was deleted from the final Statement.

Draft p. 3.1.71Issue

It should be noted that all numerical models will require more satisfactory verification on a variety of real field problems before they can confidently be applied to very long-term and large scale prediction. (218-DOI)

Response

DOE agrees with the commenter and would point out that field verification of mathematical models is an on-going effort and is an essential element in repository safety assessment.

Draft p. 3.1.71Issue

One commenter alluded to the Stone and Webster computer errors in determining piping loads during earthquakes and stated that the computer codes used in the Statement will eventually be shown to be in error. (30)

Response

Modeling of complex systems is approximate at best. Codes are continually being checked and it is hoped that the comment will not prove true to a significant extent in this Statement.

CONSEQUENCE ANALYSISDraft p. 3.1.72Issue

Discuss the relevance of the EPA assessment method cited here. (43)

Response

Discussion of the EPA Assessment method has been deleted from the final Statement.

Draft p. 3.1.73Issue

The statement--"the uncertainty of risk results from the statistical uncertainty in its models"--should be modified. (40)

Response

The statement has been deleted.

Draft p. 3.1.98 and Appendix IIssue

One commenter noted that the draft states that "...methods and detailed results for groundwater transport of radionuclides are presented in Appendix I." However, Appendix I contains no detailed discussion of groundwater transport models. That appendix is primarily a discussion of radiological consequences of leaching of waste in a repository. The hydrologic assumptions stated and presumably used in the modeling (which is not discussed) are simple (e.g., constant velocity). (208-NRC) Several commenters stated that there is no discussion of the effects of different hydrologic characteristics, i.e., no sensitivity analysis is given. (208-NRC, 218-DOI)

Response

Draft Appendix I gave references to where detailed discussions of groundwater transport were presented. The limitations of the presentation in the Statement were noted. It is believed that parameters chosen (except perhaps for Kd values) were such that the consequences presented are conservative estimates.

CONSEQUENCE ANALYSIS

Draft p. 3.1.100

Issue

One commenter noted that the probability of a meteor striking an urban area would be at least 1,000-fold greater than striking a (single) repository, based on urban population density of 3,540/km² (draft p. 3.1.145) and assuming 70% of the U.S. population in urban areas. If such comparisons are to be made, they should be reasonably accurate. (58)

Response

DOE agrees that the probability of a meteor striking an urban area is many times greater than that of a single repository based on their respective surface areas.

Draft p. 3.1.100

Issue

The destruction caused by a meteorite striking one of our large metropolitan areas is irrelevant to this consideration. We have no control over where a meteorite will fall; therefore, one place is as good as another, and the possibility of a meteorite strike does not become a consideration in the location of cities. The probability that a meteorite will disperse materials from a deep geologic repository is controllable in that the probability of a meteorite large enough to cause disruption as a function of depth and can be reduced as much as desired by going deep enough. (113-EPA)

Response

The first point regarding a meteorite strike in a metropolitan area was given simply to provide the reader with perspective. The second point regarding depth of repository is well taken and the text revised accordingly (see Section 5.5).

Issue

Several letters questioned the analysis of meteorite scenario.

Draft pp. 3.1.100 and 3.1.138--The meteorite scenario does not appear to be a credible event because there are not enough craters of sufficient size to support supposition. (35)

Draft pp. 3.1.138-147--It is not clear why a meteorite should be considered in view of the vastly greater damage the meteorite itself would do (than any dispersed radioactivity). (154)

CONSEQUENCE ANALYSISResponse

There are "astroblemes" (any scar on the surface of the earth caused by the impact of cosmic body) that have been recorded in Canada and elsewhere that suggest breach by such a mechanism is possible, although very unlikely. The meteorite scenario does serve as an upper bound for the kind of improbable events required to compromise repository containment and disperse radioactive material over large areas. Damage caused by a meteorite strike to the surrounding environment (e.g., nonradiological) are mentioned in the text as being severe, but no elaborations were made.

Draft pp. 3.1.120-123Issue

The discussion in GEIS under "routine releases of radioactive materials" does not address the problem of radionuclide contamination of ground water and run-off water. This could happen as a result of accidents, clean-up operations in storage rooms, decontamination operations during the retrieval cycle, etc.

In the section titled "Ecological Effects" seepage and water inflow from overlying strata for repositories in granite and in shale are discussed. The estimated inflow of water in a granite repository ranges from 550 to 1550 m³/day. The estimated maximum inflow during the last stages of operation in shale will range from about 3,800 to 19,000 m³/day (50,000 gpd) (sic). There appear to be two implications by omission from the discussion:

- No continued water inflow is expected in the repositories in granite and in shale after the last stage of operation.
- No water inflow is expected in the repositories in salt and in basalt.

The generic stratigraphy for salt includes possible aquifers overlying the salt bed. An area of uncertainty in state-of-the-art technology is whether the effects of mining a repository in salt and of the thermal loading are such as to create fractures that would connect the aquifer bed to the repository. TM-36/21 (p. C-1) discounts this in assuming that the permeability for salt remains at zero. No justification is provided. (208-NRC)

Response

With respect to groundwater migration of radionuclides, the impact of operational accidents and normal procedures is negligible compared to the overall repository inventory. The assumptions are that no flow will occur in any repository after final stage. The purpose of the Statement is to address the impact if flow does occur after the final stage.

CONSEQUENCE ANALYSISDraft p. 3.1.123Issue

A common fatal/nonfatal accident rate was used for surface construction activities which seems reasonable. However, a common rate was also used for the underground construction. This rate was derived from underground mining other than coal. To be closer to the truth in this area, the accident rates for more representative industries should be used. For the salt repository, the accident statistics from salt mining and potash mining should be used. For granite and basalt, underground metal and nonmetal hardrock mining is more appropriate, and for shale, use the coal mine accident statistics. (218-DOI)

Response

DOE agrees that the inclusion of this information would be desirable. However, DOE does not believe that such a level of precision is necessary nor that the conclusions of the Statement would be changed if more precise data had been available.

Draft p. 3.1.123Issue

Provide justification for all the assertions in the discussion of a tornado strike. Specifically, the dimensions of the salt pile, the size of the pieces, the probability of the tornado, its maximum wind speed, the amount of material removed and the resultant concentration in air.

In addition, no reason is given for discussing this accident. Is it the worst nonradiological accident possible, is it the only one considered, or is there another reason for its choice? What about other accidents? No conclusions are presented. Should measures be taken to protect salt piles from tornados? Has a cost-benefit analysis been made?
(208-NRC)

Response

Section 3.2 describes the basis for non-radiological accident analyses in the final Statement. Several non-radiological accidents were selected for analysis with potential for off-site consequences. The tornado, striking a salt tailings pile, is considered to be representative of non-radiological accidents at a mining site with potential for offsite consequences. The salt tailings pile at the waste repository is discussed in Section 7.4 of DOE/ET-0028. It is 1 km wide at the bottom, 910 m at the top, 30 m tall and 940 m long. All of the pile would be covered with soil except for parts that are being used. Several days mining volume could be uncovered in a pile 100 m long by 70 m wide and at 18 m high. The size distribution of salt pieces is assumed to be typical of salt in an "as mined" condition. Small sizes would be required to allow mechanical handling of the mine muck.

CONSEQUENCE ANALYSIS

No specific cost-benefit has been done at this time, but if a repository were to be constructed in salt, such an analysis could be expected.

Draft p. 3.1.125

Issue

One commenter noted that the first paragraph should acknowledge redundancy of safety devices for such an obvious hazard as a "drop" accident in a vertical shaft. (58)

Response

Page 3.1.125 of the draft Statement was intended to communicate environmental impacts due to releases of radionuclides in accidents. See DOE/ET-0028 for a complete discussion of accident scenarios.

Draft p. 3.1.125

Issue

Uncertainties in doses predicted by models are misleading. The discussion should indicate the uncertainty to be expected when a critical parameter has a range of values such as the magnitude of earthquake, floods, etc. (208-NRC)

Response

The Statement calculates the dose caused by a natural phenomenon capable of producing the given effect. The range of possible phenomenon intensity is not included.

Draft pp. 3.1.136-172

Issue

In this section several scenarios resulting in the release to the biosphere of large amounts of radioactivity are postulated. Because of the generic nature of the repositories and the lack of specific data needed in the calculation, many of the parameters controlling the physical transport of the radionuclides are not even known to order of magnitude certainty. The resulting dilutions that are used in the dose models have even larger bands. Therefore, breaking down the resulting doses by reprocessing procedure and rock type makes little sense, when the differences between them are much less than the error band due to transport-dose modeling. (208-NRC)

Response

The DOE agrees with the issue statement. One useful purpose of the analysis as presented is to show differences between alternatives for "all else being equal". It also

CONSEQUENCE ANALYSIS

demonstrated the point suggested, e.g., that the differences between alternatives in terms of radiological impact are insignificant compared to other considerations and uncertainties.

Draft pp. 3.1.136-172

Issue

Section 3.1.5.2 is entitled, "Potential Impacts Associated with Repository Wastes in the Long-Term." Although this section gives population doses caused by different accident scenarios, it does not discuss the problem of land contamination caused by these accidents. (208-NRC)

Response

Land contamination is included in the sense that part of the population doses is derived from crops grown in the contaminated land.

Draft pp. 3.1.136-172

Issue

The section on long-term impacts is devoted entirely to accidents that may breach the repository, most of which are presented as being so improbable that they are unlikely to even occur. No discussion is presented of expected long-term impact. If the facility is sited, filled and sealed according to plan, what will the long-term consequences of this action be in the absence of unlikely accidents? This question is discussed partially in Appendix I, but the discussions are not presented in the text of the GEIS as projected impacts of the action. (208-NRC)

Response

If the repository acts as planned no long-term consequences will occur (aside from small amounts of uplift and subsidence).

Draft pp. 3.1.136-172

Issue

Several letters commented on the selection and classification of accident scenarios analyzed in the draft Statement.

A different classification system than the one used for the long-term accident scenarios in the Statement should be considered. (198)

Releases are estimated for four hypothetical accident sequences. The numbers associated with the releases are presented by the GEIS as "what if" calculations, without

CONSEQUENCE ANALYSIS

discussion of why these sequences are important except to say that they are "believed most representative" of release events. How these events were chosen and why they are believed to be representative and to bound the impact of long-term consequences should be discussed. (208-NRC)

Response

As described in DOE/ET-0028 and DOE/ET-0029, an umbrella source term concept was used to limit the number of accidents requiring detailed impact analysis. Viewed independently of accident initiation sequences and frequencies, the accident source terms can be grouped by release severity for environmental consequence analyses. Release groups were defined based on similar release pathways, chemical form, accident severity, and isotope types released. The umbrella source term was that accident having the largest release in a particular group of accidents.

The information below lists those postulated accidents (for geologic repositories) examined in this Statement and notes the umbrella source term.

<u>Minor</u>	<u>Umbrella Source Term</u>
LLW drum rupture due to handling error	
Minor canister failure	X
Receipt of externally-contaminated canister	
Dropped shipping cask	
<u>Moderate</u>	
Waste container drop during handling	
Waste package dropped down mine shaft	X
Tornado strikes mined materials storage area	X
LLW drum rupture due to mechanical damage and fire	
LLW drum rupture due to internal explosion	
<u>Non-Design Basis</u>	
Nuclear warfare	
Repository breach by meteorite	X
Repository breach by drilling	X
Repository breach by solution mining	X
Volcanism	
Repository breach by faulting with groundwater transport	X
Erosion	
Criticality	

CONSEQUENCE ANALYSIS

The following definitions of minor, moderate, and non-design basis were used.

- Minor - infrequent events with potential for small material release, major equipment damage, or the creation of radiation fields in occupied zones which could result in occupational exposures exceeding 10 CFR 20 limits (5 rem/yr).
- Moderate - relatively frequent occurrences involving process interruptions with potential for significant release of radioactive or other hazardous materials.
- Non-design basis - events which exceed site criteria.

Site criteria include 1) definition of the maximum credible earthquake, surface faulting, floods, and wind velocities based on historical evidence, local and regional geology, and expert judgement; 2) local and regional demography; and 3) proximity and definition of hazards caused by man.

Draft p. 3.1.137

Issue

The groundwater releases did not appear to include the "two aquifer case," which is most significant for groundwater releases. This case involves a hydrologic connection between two aquifers through a repository, with subsequent groundwater transport. The analysis also appears to consider only release of the total radionuclide content, which does not appear to be a credible or useful form of analysis. (113-EPA)

Response

Such a scenario was added to the final Statement (see Section 5.5).

Draft p. 3.1.137

Issue

Line 7 is the first mention of phosphates. What is the tie-in with salt repositories? (218-DOI)

Response

Phosphates are often found in association with salt deposits as both are the result of evaporate deposition. See also p. 3.1.8 of draft.

CONSEQUENCE ANALYSISDraft p. 3.1.138Issue

References relevant to this discussion and not cited include:

1. K. A. Solomon, R. C. Erdmann and D. Okrent, "Estimate of Hazards of a Nuclear Reactor from the Random Impact of Meteorites," Nucl. Technical, 25. 68 (1975)
2. K. A. Solomon, R. C. Erdmann, T. E. Hicks and D. Okrent, "Estimates of the Hazards to a Nuclear Reactor from Random Impact of Meteorites," USCL-NEG-7426, University of California at Los Angeles (March 1979). (208-NRC)

Response

These were reviewed, but not included in the reference list.

Draft p. 3.1.138Issue

The presence of salt would probably not preclude the use of the water as a source of food or recreation. The salt would be diluted to acceptable levels by any reasonable amount of water far more quickly than the radioactivity. (113-EPA)

Response

In the faulting and flooding accident large quantities of water flow rapidly through the repository and then to the surface. Once at the surface these waters flow to a larger river in the area. If the assumption is made that salt would not interfere with use of the stream for food and recreation, then the non-salt repository data are applicable. However, the DOE feels that the scenario is reasonable in that salt content would preclude the use of the stream.

Draft pp. 3.1.138-147Issue

This appears to be based on the Gera, Claiborne paper which contains an error of 10,000. Whether the draft EIS picked up this error is not clear. We have been unable to reproduce the draft EIS results. Our results are lower than the draft EIS values at 2,050, 100,000, and 1 million years but higher at 1,000 years. (154)

CONSEQUENCE ANALYSISResponse

The Gera, Claiborne paper was not used in a manner that would make any errors in its data applicable to the radiation doses incurred by the public as a result of this hypothetical accident.

Draft p. 3.1.139

Issue

What are the bases for assuming that 10 percent of the particulates suspended are of respirable size? (113-EPA)

Response

Based on the explosive nature of the meteorite impact, a conservative estimate of 10% of airborne material in the respirable size range ($<10 \mu$ AMAD) was assumed. This assumption is based on the expected size distribution of all material that becomes airborne as a result of the meteorite strike. Most of this material will be larger pieces of debris.

Draft p. 3.1.147

Issue

A meteorite of the described size would undoubtedly produce a local disaster area. The impact of the meteorite, however, would also disperse radioactive materials into the atmosphere from which they would impact over an extended area. The additional impact of this radioactive material is what is significant. The impact would not likely be local nor be controlled by local monitoring. (113-EPA)

Response

The dose to the population of the eastern U.S. was calculated and presented (see Section 5.5).

Draft p. 3.1.147

Issue

Even for an incredible scenario in which water enters the repository the year it is closed (2050), goes unnoticed, works its way 10 km to a stream where it is used in all possible ways and for which conservative assumptions were used for every parameter, it is shown that society's risk from lightning is 10 million times the risk from this event! At 1,000 years the probability that such a release would go unnoticed is somewhat higher. If this event were to occur at 1,000 years, maximum individual doses are 5 to 10 mrem/year and population doses are about 1% of natural background.

CONSEQUENCE ANALYSIS

This highly important section, if read very carefully, supports the Utility Waste Management Group position - namely, a most conservative bounding calculation of the results of repository breach shows that there are no catastrophic doses. This fact is obscured, however, by

- Simply poor presentation.
- Using 50-yr and 70-yr accumulated doses, the reader sees a relatively larger (70X larger) number than he may be used to thinking of (100 mrem/year background; 3 rem/year occupational dose).
- Going overboard on the bounding. For example, in addition to the somewhat plausible cases of 0.1%/yr and 0.01%/year leaching, cases for the entire (100%) leaching in a single year. Now that is truly incredible and there is no known mechanism by which this could happen. All these calculations are put together in the same tables. The only things people generally are likely to credit are the largest values.

We suggest that the final EIS presentation could be vastly improved by

- Sticking to first-year maximum individual doses with an indication on each table of the comparison to natural background and occupational dose rates.
- Separating the more plausible calculations (unnoticed breach after a few hundred to a thousand years and leach rates in the range of 0.1 to 0.01%/year) from the implausible "bounding calculation" (immediate breach and 100% leached in one year). (154)

Response

The presentation of the data was clarified and simplified in the final Statement. The 50 year doses referred to were adjusted to reflect 70 year-estimates. The bounding condition of 100% leaching in one year was removed for the final Statement.

Draft p. 3.1.148

Issue

The scenario does not appear to be a particularly bad case because of the limitation of the contact for one year. Such a limitation, together with slow leaching, results in a minimal release of radionuclides. What is the effect of continued erosion? (113-EPA)

Response

Final Table 5.5.9 gives 70-yr accumulated dose from continued leaching.

CONSEQUENCE ANALYSISDraft pp. 3.1.148-155Issue

Discuss the reasons for the choice of $2.8 \text{ m}^3/\text{sec}$ (100 cfs) for water flow through the breached repository. Identify the flow rate of hypothetical River "R" used in transport and dilution calculations. (208-NRC)

Response

The choice of $2.8 \text{ m}^3/\text{sec}$ (100 cfs) was arbitrary. The flow of River "R" is $120 \text{ m}^3/\text{sec}$.

Draft p. 3.1.149 and 3.1.155Issue

The leach rates presented on p. 3.1.149 require further examination. The long-term leach resistance of the spent fuel is $1 \times 10^{-5} \text{ g/cm}^2/\text{day}$ and times a fracture factor of 5 equals $5 \times 10^{-5} \text{ g/cm}^2/\text{day}$ which is lower than the $10 \times 10^{-5} \text{ g/cm}^2/\text{day}$ for vitrified high-level waste. There is no physical evidence to support a lower leach rate for spent fuel. (198)

Response

The leach rate figure of $10 \times 10^{-5} \text{ g/cm}^2/\text{day}$ presented for vitrified high-level waste was a typographical error. The information in the table should have been presented as follows:

<u>Leach rate</u>	<u>Time</u>
$1 \times 10^{-4} \text{ g/cm}^2/\text{day}$	first 10 days of leaching
$1 \times 10^{-5} \text{ g/cm}^2/\text{day}$	remainder of time

See also response below.

Draft p. 3.1.149Issue

A leach rate of $10^{-4} \text{ g/cm}^2/\text{day}$, applied to a one centimeter cube of density 2, would result in a leaching rate of 3×10^{-4} per day, or approximately 0.1 per year. Is this the value that was used in the analysis? (113-EPA)

CONSEQUENCE ANALYSISResponse

Actual leach rates for HLW in glass used were

1×10^{-4} g/cm²/day - for first 10 days of leaching

1×10^{-5} g/cm²/day - for remainder of time

This is equivalent to approximately 1% loss of material contacted per year by leaching assuming a density of waste in glass as 3 g/cm³.

Draft p. 3.1.149Issue

What is the source of the listed leach rate (1×10^{-4} gm/cm²/day) for intermediate level and low-level waste? What waste forms are involved? (218-DOI)

Response

The values listed for intermediate-level and low-level wastes are unpublished consensus judgments of senior individuals at Battelle-Northwest Laboratory. Another individual believed that 1×10^{-3} gm/cm²/day was more appropriate for wastes solidified with cement. In terms of radiological impact intermediate-level and low-level wastes are not significant in comparison with high-level wastes.

Draft pp. 3.1.150-155Issue

The annual doses to a maximum individual associated with the breach of a salt repository are three to ten times the permissible annual dose for occupational exposures... Thus the calculated number of health effects attributed to this accident would range from 1×10^4 to 3×10^5 .

GEIS goes on to multiply these figures by 1/100 as the probability of failure of waste containment and by 4×10^{-11} /yr as the probability of a new fault intersecting the repository to arrive at insignificant risk levels. The probability of an existing fault becoming permeable should also be considered. (208-NRC)

Response

The case presented was felt to be sufficient; site-selection criteria are expected to preclude the presence of existing faults. (However, see also EPA's comments regarding probability of interaction with water).

CONSEQUENCE ANALYSISDraft p. 3.1.155Issue

Values for southeast New Mexico should not be used in a generic Statement. Certainly faulting rates of 10^2 /yr typical of the San Andreas system could be avoided. A reasonable, "conservative" upper bound for this analysis might be 10^{-4} /yr. The discussion on p. 3.1.156 would then conclude that the risk from repository breach by faulting and flooding would be no greater than the risk from lightning, assuming the rest of the analysis is correct -- not definitely seven orders of magnitude. (218-DOI)

Response

DOE assumes the commenter was referring to a faulting rate of 10^{-2} /yr (as noted in the draft) as opposed to 10^2 /yr. In addition, DOE would agree that the rate used is an ultra conservative one. However, the consequence analysis is generally based on such conservatism. The fault rate identified was documented in the draft Statement as referring to work done in the Delaware Basin.

Draft p. 3.1.157, Table 3.1.42Issue

Dose commitments of 10^8 person-rem are estimated to result in 2×10^4 fatal cancers. The risk associated (including the probability) is much less. (113-EPA)

Response

The DOE agrees that the risk associated with the accident (including probability) is much less; this is addressed in Section 5.5.

Draft p. 3.1.158Issue

Provide a reference for ten dilution factors given and discuss the cause of the 50-fold differences shown. (208-NRC)

Response

The data were derived from Hanford Annual Environmental reports, which represent several years data. The differences in dilution factors are due to the adsorptive qualities of certain elements when in contact with suspended river sediment. Those elements with dilution factors of 100 are deposited in river sediment and do not reach the estuary. Others such as ^3H , reflect only the increased river flow rate between the release point and the estuary.

CONSEQUENCE ANALYSISDraft pp. 3.1.160-161Issue

One commenter noted that the special parameter choices for migration length and ground water velocity would effect the calculations. When all waste is leached in one year (in the faulting scenario) the assertion that the 50-year cumulative total body dose is at most 1 to 3 times occupational limits does not make sense. (40)

Response

The consequence analysis of groundwater transport employed a very restricted set of parameters. This was not by choice but what was available to use within the time allowed.

The scenario of all waste leached in one year was removed.

Draft p. 3.1.161Issue

The release rates used in the groundwater transport analysis in the main body of the report jump strangely from 100%/yr to 0.1%/yr. The base case should clearly be for the low rates, but results for intermediate release rates should be presented. The consequence analysis should consider solubility limits of the various radionuclides in the groundwater system under consideration; the present analysis assumes varying source terms for nuclide transport, some of which may not be physically possible. (218-DOI)

Response

The bounding condition of 100% leaching in one year was removed in the final Statement (see final Section 5.5).

Draft p. 3.1.162Issue

Distribution of the waste would lower the maximum and regional individual doses, but would increase the probability of the event by a factor equal to the number of repositories. (113-EPA)

Response

DOE is in agreement with the comment; the mathematical expectation of societal risk would remain unchanged.

CONSEQUENCE ANALYSISDraft pp. 3.1.165-168Issue

Several letters noted that the repository breach by drilling accident assumes that drillers working in an environment where salt could be expected, and who are looking for salt, will assay samples which are not salt. If drillers are looking for salt, they do not need an assay to determine its presence and will not burden themselves with the additional cost of assays. This entire section then is based on an assumption which is not realistic. (114, 214)

Response

While it may be unnecessary for drillers to determine that they have reached salt, analysis would probably be required before purification would begin.

Draft pp. 3.1.168-172Issue

One commenter noted that the solution mining accident is limited in scope and duration and assumes people 1,000 years from today possess today's radiation technology. (114)

Response

The final Statement notes (in Section 5.5.5) that other solution mining calculations have obtained results of somewhat larger doses than those presented in this Statement. Although possible, it is not believed realistic that people living 1000 years from today will possess technology equivalent to or in advance of today's.

Draft pp 3.1.168-172Issue

One commenter noted that the consequences of solution mining of contaminated salt are stated in terms of man-rems. More detailed information should be given as to whether this is a whole-body dose, thyroid dose, etc. (166)

Response

In each of the accidental breaches of a repository, the whole-body dose was calculated and the health effects postulated to result were given.

CONSEQUENCE ANALYSISDraft pp. 3.1.168-172Issue

Even on the basis of incredibly pessimistic assumptions the accident caused by solution mining in a salt repository is shown to have negligible consequences. In this case we find somewhat higher values than the draft EIS. Is it possible that the decontamination factor referred to on p. 3.1.171 was applied? (154)

Response

The decontamination factor was not used.

Draft pp. 3.1.179-192Issue

The potential hazards of storage and transportation of radioactive waste did not include justification of the accident frequencies (the number of occurrences over a given time period) that were used. (40)

Response

Basis for the analysis of predisposal activities, including accident frequencies, are detailed in the support document DOE/ET-0028. DOE/ET-0054 (DOE 1978a) also contains an analysis of storage and transportation of spent fuel.

Draft p. 3.1.200Issue

Several commenters noted that more complete analysis should be performed on the potential for accidents during storage and transportation of PuO_2 .

Statement is made that "no accidental release of radioactive material is postulated for shipments of plutonium oxide." DOE has achieved perfection. No accidents with plutonium oxide in transportation will occur. (30)

There is a very brief discussion of a criticality accident in a storage facility for plutonium oxide. If anything, this treatment is even less convincing than the others. The suggestion is made that 200 grams of stored plutonium oxide would reach the atmosphere in the worst criticality accident. As you know, this is a very large amount, biologically speaking. But the progress of such an accident is never discussed. Rather it is claimed that the maximum 70-yr total body dose commitment to any individual would be 75 millirems. Similarly, these paragraphs imply that no accident analysis has been done for a surface storage facility for plutonium oxide. (40)

CONSEQUENCE ANALYSIS

Note the assumption that "no accidental release of radioactive material is postulated for shipments of plutonium oxide." (154)

Response

If a future national policy decision leads to the reprocessing of spent fuel and the recovery of PuO_2 , PuO_2 will not be considered a nuclear waste but a resource.

The impacts of handling PuO_2 have been considered by the Department and its predecessor agencies in other environmental statements (ERDA 1976a) and by the Nuclear Regulatory Commission in its proceedings on GESMO (NRC 1976a). Further consideration of the impacts of PuO_2 storage and transportation would be covered in any reopening of the GESMO proceedings. Recognizing the in-place mechanisms to ensure radiological safety and safeguards, no credible accidental release of PuO_2 was envisioned.

Draft p. 3.1.212Issue

On p. 3.1.212 is the apotheosis of this EIS approach to risk. It is asserted that in case of theft or sabotage, the risk to society will be small if any of the contributing probabilities or consequence are small. This is a revealing claim. Just such logic has always been used in evaluation nuclear risks over the last 20 years. What is implicitly being asserted is that no matter how large the consequences of a particular event may be, if the probability is sufficiently small the risk to society is negligible. It may be that society should take that view, but those who believe in it should be prepared to offer justification (including definitions of "small" and of what constitutes an appropriate estimation procedure). (40)

Response

The definition of risk used in this Statement is the probability of an event times the consequences of that event. Using this definition, events with relatively large consequences and small probabilities have low risk. Mathematically low risk level is not meant to imply risk acceptability. This is an issue to be decided in the regulatory and public hearing process. The revised Statement format includes a section on the discussion of risk perspective. See Section 3.4 for comments on risk.

Draft Appendix IIssue

Appendix I discusses the possibility of release of radionuclides to the biosphere through groundwater mass transport. The impression given is that container life will be

CONSEQUENCE ANALYSIS

about 1,000 years and that no significant release is expected for one million years. This is in apparent contradiction to results given in TM-36/21 (pp. xiv, 8.5 and 8.6). What is the expected rate of corrosion of the canister and the sleeve in salt brine or in fresh water? What are the values (or ranges) of effective hydraulic conductivity, porosity, retardation factors and hydraulic gradients of the rock mass surround the repository that were used to obtain Tables I.1 to I.12? (208-NRC)

Response

This was intended to be a parametric analysis and actual container life was not given. The commenter may have misunderstood the term "release". In some cases, no significant releases to the biosphere may occur for a million years.

Draft Appendix I

Issue

This appendix is deficient. It is based on leaching of the entire repository by ground water, passage of nuclides through a rather freely flowing aquifer, and discharge into a large surface stream (10,000 cubic feet per second or 8.9×10^{12} liters per year). If we apply the generic density of population in terms of river flow from our forthcoming dose assessment report, which is 3.3×10^{-7} person years per liter, the river is capable of being a water supply for about three million people, a great many of whom would receive close to the maximum individual dose. (113-EPA)

Response

The R river in the reference environment supplies about two million people which is close to the three million as estimated above using a generic population density. The drinking pathway is the only pathway of exposure. In the dose distribution a large fraction of the down stream population might receive nearly that dose received by the maximum individual. In another groundwater scenario a ratio of 1/5 was determined between the maximum individual and the per capita dose.

Draft Appendix I

Issue

An apparent conflict exists between the basic assumptions in the main text and Appendix I. The main text stated that "...disposal in salt has been emphasized..." (p. 3.1). However, the assumptions made in Appendix I (p. I.9) for an earlier analysis (which was the basis of the current version of the Impact Statement) assumed that the repository is in a non-salt media. Furthermore, some of the details of the model should be briefly summarized in the appendix. The statement "Detailed descriptions of these models

CONSEQUENCE ANALYSIS

are found in references 1-7" (p. I.4), is not sufficient. GETOUT (p. I.10) should be briefly discussed as well. (113-EPA)

Response

There is some confusion as to the basis for the analysis in the Statement. The material of Appendix I was extracted from an earlier report but the methods of Appendix I were used for the slow groundwater transport analysis. No emphasis on geologic media was intended. Contamination of an aquifer by salt may preclude use of the aquifer for domestic purposes.

Draft Appendix IIssue

The criterion for public acceptability of 120 millirems per year to the maximum individual is not defensible, and population dose needs primary consideration. If the approximately three million people who could be supported by the river were each to receive 120 millirems per year, the population dose would be 350,000 person-rems per year or approximately 70 health effects per year using the BEIR-whole body estimate. Although all the postulated three million people would not receive the maximum individual dose and although these nuclides would not produce whole body doses, there is no reason to believe that the concentration of nuclides in the river would decrease substantially as the nuclides moved down river. (113-EPA)

Response

Although the suggested scenario results in 70 health effects/yr which relates to 7×10^5 health effects over 10,000 years the probability (using EPA's value of 4×10^{-7}) reduces the mathematical expectation of societal risk of less than one health effect over 10,000 years. Regardless, the criterion stated was developed for a referenced report. DOE agrees that population dose from the events considered here should be controlling.

Draft Appendix IIssue

The analysis uses unquoted sorption equilibrium constants. These are probably the Battelle desert soil values which may be unreasonably high. The text on p. I.6 refers to "three miles of western U.S. subsoil," which is reminiscent of the Battelle "desert soil." These sorption constants are not necessarily typical of all soils and rocks and, in any case, should be listed in tabular form. (113-EPA)

CONSEQUENCE ANALYSISResponse

References at the end of final Section 5.5 contain the information the commenter requests.

Draft Appendix IIssue

Just as it is improper to neglect population dose in the river, it is improper to neglect individual dose to users of groundwater. This is completely omitted in this section. Since the ground water velocity is stated to be equal to one foot per day, or a little over 100 meters per year, the aquifer would be expected to be a good water provider and comparable to the aquifer of draft Appendix F which is stated as supporting "numerous shallow wells supplying residences and farms" and also a "public water supply well" for a city. Population dose from use of the aquifer may very well be significant in addition to individual doses. (113-EPA)

Response

An additional case involving a well intersecting a contaminated aquifer has been added (see Section 5.5).

Draft Appendix IIssue

We believe the comprehensive model used in the safety analysis is not applicable on a generic basis. The modeling efforts of H. C. Burkholder and his colleagues at Battelle are pioneering and commendable. However, in Appendix I the assumptions used in the model analysis are clearly spelled out on page I.9. Among these assumptions are a) that "the repository is located in a non-salt formation surrounded by a geology with nuclide retention properties similar to those for a particular Hanford Reservation subsoil"; and b) "the groundwater flows into a surface stream with a flow rate of 10,000 ft³/sec (1/10 the flow rate of the Columbia River near the Hanford Reservation) where the nuclides are further diluted." This flow is equivalent to the average flow of the Delaware River at Trenton. With these and other simplifying assumptions, the model predicts a benign outcome. However, the problems are multiple.

First, although dilution of the radionuclide-bearing groundwater by a 10,000 ft³/sec river is one plausible scenario for radwaste dissolved in Hanford ground waters, a 10,000-fold concentration might occur in other environments, for example, in areas where groundwater flow is toward marshes or wet playas. Second, what is the dose to man if the groundwater were tapped by a future town well-field upgradient from discharge into the river? Third, the Kd values for Hanford subsoil are unlikely to be applicable to fractured media.

CONSEQUENCE ANALYSIS

Briefly, the model is acceptable for one HLW scenario in Hanford alluvium. It is unacceptable for other scenarios at Hanford, and certainly unacceptable for any other rocks and waste types. Therefore, the seemingly comprehensive tables comparing health effects from radwaste disposal in salt, granite, shale, and basalt are difficult to justify. The draft EIS itself in several places follows the IRG in emphasizing the importance of site specific studies. We suggest the presentation of considerable numerical data in Section 3.1.5.2 is not warranted; this should be resolved in the final Statement. (218-DOI)

Response

Modeling of groundwater migration is indeed limited because of the choice of parameters. As suggested, less emphasis has been put on the analysis. An attempt to circumvent this problem was the introduction of a flooding and large stream flow scenario. This too is of limited use. In addition, a contaminated well scenario has been added.

Draft p. I.1

Issue

One commenter raised questions on the following statement: "...the geologic repository system provides two potential means of protection from radioactive waste. The first means is containment of the waste for a sufficient length of time for the hazardous nuclides to decay to innocuous levels such that unrestricted release to the environment presents no radiological hazard. The second means is to limit the rate of release of nuclides to the biosphere such that their concentration in the constantly renewed local surface water and air never exceeds acceptable limits. The actual repository system will provide protection using some effective balance of these two means."

What do the terms "innocuous levels," "acceptable limits," and "effective balance" really mean? These should be quantitatively defined. This statement indicates that a geologic repository system may be designed to include planned releases to surface water and air. Evidently the authors still support the archaic idea that the solution to pollution is dilution. (97)

Response

Such expressions as "innocuous levels," "acceptable limits" and "effective balance" have been, by and large, removed from the final Statement in favor of more meaningful language.

CONSEQUENCE ANALYSISDraft p. I.2Issue

The discussion of compensating for a poor site by an extremely durable waste container in the last paragraph of page I.2 is irrelevant, since human intrusion cannot be ruled out. (113-EPA)

Response

The paragraph cited relates only to compensation in outcome from one release scenario as a function of variation in parameters.

Draft p. I.3Issue

The concentration on individual dose rather than population dose is again shown in the fourth paragraph on p. I.3 which speaks of reducing the ^{129}I dose by a factor of 10 by reducing the release rate by a factor of 10. Population dose would not be changed. (113-EPA)

Response

The analysis dealt with in Appendix I and publications in which it was based concerned the maximum individual. It is agreed that population doses should be considered and in the reference statement the collective population dose would not be changed.

Draft p. I.3Issue

The leach rate figures used throughout and specifically in Figure I.3 are unrealistically low. The "hypothetical waste management system characterization" is about a factor of 10 better than the values we have been given by our consultant, Arthur D. Little, Inc., and contrast strongly with the estimate of the EPA geologist panel: "There is no evidence that incorporation into a glass will ensure resistance to significant leaching over time scales over a decade." (EPA 520/4-78-004 page 7). (113-EPA)

Response

The referenced EPA report, EPA 520/4-78-004, by the Arthur D. Little, Inc., seems to disagree with other investigators in the field. Furthermore, the values in draft Figure I.3 are based on leach data obtained in the lab and in the field which appear to span the

CONSEQUENCE ANALYSIS

rates to be expected in a repository. One of the possible shortcomings of the values in draft Figure I.3 is that they are determined for unfractured monolithic cylinders of the various glasses.

Draft p. I.7

Issue

Draft Figure I.4 appears to require a leach time of 100,000 years for "satisfactory" (less than 120 millirems per year) operation. This may not be possible for all contained nuclides, since some nuclides are geochemically mobile. (113-EPA)

Response

A confusion of terms appears in this comment. If the commenter refers to time when a nuclide is discharged to the biosphere, then a 100,000-yr water travel time is not unrealistic. By dating methods groundwater has been shown to be millions of years old.

Draft p. I.10

Issue

The notion that the dose from ^{226}Ra can be reduced by limiting the leaching of ^{238}U is incorrect. That ^{238}U migration could be controlled over its half-life (4.5 billion years) is doubtful.

We believe that the impact analysis is in a premature stage in this section. The analyses stated in Appendix I are divided into two categories: past work and present work. Since the present work is only partially complete, the results presented in the Statement may be revised when the present work is completed. This may change results in the stated conclusions in the Statement. We believe the present work should include an error analysis and sensitivity analysis.

All the references to this appendix are from Battelle Pacific Northwest Laboratories work. Has any of this work been performed elsewhere? (113-EPA)

Response

There was no suggestion in Appendix I that ^{238}U could be contained for 4.5 billion years or more. Radon-226 is a relatively short lived daughter of ^{238}U . In many systems Ra is less mobile than U. Furthermore, in most cases analyzed up to date, Ra is released to the biosphere only because it is a daughter of a long-lived parent. Therefore, the Ra dose can be very sensitive to ^{238}U leach characteristics.

As noted in final Appendix L much of the work done with respect to nuclear waste management has been conducted by Battelle Pacific Northwest Laboratories (PNL). With respect

CONSEQUENCE ANALYSIS

to nuclide mobility in geologic media, again much of the work has been either conducted by or subcontracted by PNL.

Draft pp. I.11-15Issue

The assumptions used in formulating the computer models and the variables utilized to generate the data should be fully explained and qualified. (97)

Response

DOE does not feel that such information is appropriate for inclusion in this Statement because of bulk considerations.

Draft Appendix LIssue

One commenter questioned whether accidental releases by means other than sabotage had been considered. (113-EPA)

Response

Releases from high level liquid waste solidification and fuel residue packaging facilities are not limited to those of sabotage. Operational accidents can release small amounts of radioactive material to the environment. Accidents for predisposal waste treatment operations are discussed in Section 4.8 of this final Statement.

Draft Appendix MIssue

The accidents leading to releases of radionuclides (Tables M.3 and M.8) are not characterized, so it is impossible to understand what is involved. The basis for release of 0.1% of total ^{85}Kr (p. M.52) is not given. The total releases of 22 Megacuries of ^{85}Kr should be compared with the permissible 40 CFR 190 values. There is no consideration of possible radionuclide releases from accidents in a spent fuel storage facility in Table M.52. There is some discussion in Table M.61 but there is no basis for judgement as to the releases or selection of accidents. For example, there is no discussion of the effect of loss of coolant in water basin storage through failure of the tank or through sabotage.

Note also that the risk estimates, pp. M.6, M.33, M.53, M.81, M.87, etc. will require revision if numerical risk coefficients are changed since all are derived from the risk coefficients developed in Appendix E. (113-EPA)

CONSEQUENCE ANALYSISResponse

The format for this environmental impact statement does not include a detailed description of potential accidents for each waste treatment and storage technology. Individual accidents are described in DOE/ET-0028. Section 5.1 of that report discusses the storage of high-level liquid wastes. Section 5.2 discusses the storage of fuel residues.

The release of ^{85}Kr discussed on page M.52 of the draft Statement is due to anticipated leakage from the pressurized storage cylinders. Large cylinder failures also contribute to this number. The total ^{85}Kr release of 2.2×10^7 curies over 30 years is less than that allowed in 40 CFR 190. Up to 50,000 curies per Gigawatt year of electrical generation is allowed in that regulation.

Accidents related to spent fuel storage are discussed in Section 4.8 of this final Statement. Further discussion is given in Section 5.7 of DOE/ET-0028. The basis for accident analyses in this Statement is discussed in final Section 3.2. Sabatoge was not considered in the development of accident scenarios. Credible loss of coolant accidents in the water basin storage of spent fuel, including tank failures, would result in no significant releases to the spent fuel facility.

Draft Appendix NIssue

The largest accident consequences presented in the GEIS occur during the transportation of radioactive wastes. Much of the detailed analysis is contained in DOE/ET-0029 as stated. An examination of these two documents reveals that accident release fractions, curie amount of isotopes that may be released, and doses to affected individuals are provided. However, some important details concerning accident assumptions are not given. These detailed assumptions involve the fraction of released material that is aerosolized in respirable form. Also missing are resuspension factors. In Appendix B, DOE/ET-0029, reference is given to other reports and computer codes that may contain these factors. These assumptions need to be outlined directly in DOE/ET-0029 so that factors in the degree of realism of the accident analysis can be more easily evaluated and the conclusion compared to other study results. (208-NRC)

Response

It was assumed that 100% of the released material was airborne and respirable. Resuspension factors were not used. The release was assumed to last the length of time given for each accident. Released material was calculated directly as the product of release fraction times inventory. Additional information can be found in DOE/ET-0028.

CONSEQUENCE ANALYSISDraft Appendix NIssue

Our comparison of the impacts presented in the GEIS with those in DOE/ET-0029 examined spent fuel shipment only. In converting results from one document to another, several errors have apparently been made. The remaining transportation sections in Appendix N should be similarly reviewed for errors. (208-NRC)

Response

Errors in impacts presented in Appendix N were found. These errors were corrected for the final Statement.

Draft Appendix NIssue

Throughout Appendix N, the total body radiation dose from the routine transport of radioactive materials is given in various tables. These tables show the dose to the population residing along the transport route and to members of the transport work force. The tables omit the dose to occupants of vehicles using the same route in the case of truck transport. It is not clear whether the dose that results from a delay in transit of the radioactive shipment has been included. These delays could occur from a traffic jam or a stop at a truck stop in the case of truck transport. For rail transport, a delay can be caused by adverse track conditions or a mechanical breakdown. (208-NRC)

Response

Vehicle occupant doses are given in each appropriate section of DOE/ET-0029. Transit delays are accounted for in the assumed average velocity of each shipment. Doses in man-rem are unchanged if the vehicle moves in spurts or continuously as long as the total travel time is unchanged.

Draft p. N.4Issue

The transportation accident consequences presented on p. N.4 of the GEIS are based on accident number 6.2.8 described in Table 6.2.6 of DOE/ET-0028. Releases of cesium are based on vaporization mechanisms as reported in Supplement II to WASH-1238. A study conducted by Battelle's Pacific Northwest Laboratory, "An Assessment of the Risk of Transporting Spent Nuclear Fuel by Truck," PNL-2588 indicates that other mechanisms can cause additional releases of cesium and other isotopes. These mechanisms involve either oxidation or leaching of the fuel. Releases of radioactive material resulting from these mechanisms can occur

CONSEQUENCE ANALYSIS

in addition to the releases used in accident number 6.2.8. The probability of accidents occurring where several release mechanisms operate is less than the probability associated with accidents where only a few release mechanisms operate. Thus the risk may be greater for the latter accident than the one involving many release mechanisms. Recommend the GEIS address these accidents that involve several release mechanisms and show that either the risks involved are less than those of accident number 6.2.8 or if the risks are greater, this more severe accident should be used as the umbrella source term for severe accidents. (208-NRC)

Response

An Assessment of the Risk of Transporting Spent Nuclear Fuel by Truck (PNL-2588, Elder et al. 1978 in reference list) was not available at the time of the writing of DOE/ET-0028. Available literature at the time, primarily WASH-1238 (AEC 1972), stated that a vaporization release of cesium was the most likely mechanism for the release of radionuclides from a spent fuel accident of the type considered in this EIS. Other mechanisms for radionuclide release may exist, but they are either small compared to those used in the present Statement or require cask failures in excess of the design basis accident spectrum. PNL-2588 (Elder et al. 1978) and DOE/ET-0028 agree that cesium is the largest contributor to consequences from accidents involving overheated fuel with a small breach of the cask wall or penetration.

Draft p. N.4

Issue

Although the radiation dose to the maximum individual from postulated accidents are given, the total population dose to persons in the vicinity of the accident is not given. Since this is an important environmental impact, it should be included in the GEIS in context with accident frequencies.

The actual value for this population dose can be found on p. 4.1.10 of DOE/ET-0029. The 70-year dose commitment is given as 140 man-rem. Although the analysis uses a population density of 90 persons per square km for routine radiological impacts, the population density used for the accident analysis is 130 person per square km. Note that population densities in suburban or urban areas can be at least an order of magnitude higher than this population density. A severe accident occurring in a suburban or urban area would, therefore, have a substantially greater environmental impact than the accident consequences presented in the GEIS. In order that all relevant impacts be included in the GEIS, recommend including the consequences of severe accidents in high population density areas.

The largest accident dose reported in the GEIS results from a severe accident involving a rail shipment of spent fuel. The resulting whole-body dose to the maximum individual is given as 120 rem for a one year period following the accident. The dose is based on the

CONSEQUENCE ANALYSIS

amount of radionuclides released to the atmosphere as given in Table 4.1.1-12 of DOE/ET-0029. The amounts given in this table are based on release fractions given in Table 6.2.6 for accident number 6.2.8 in DOE/ET-0028. An examination of the release fractions and cask inventories given in DOE/ET-0028 indicates that the amount of radionuclides given in DOE/ET-0029, and hence the dose reported in Appendix N, are in error. There are three sources of error. Mixed fission products and actinides have been excluded from the release, the amount of ^{85}Kr released is underestimated, and the amount of ^{137}Cs released has been overestimated.

Finally, the following discussion shows that the amount of ^{134}Cs and ^{137}Cs released for accident number 6.2.8 has been overestimated. The discussion on page 6.2.14 of DOE/ET-0028 indicates that 6×10^{-4} of the cesium inventory may be available for release as a result of fuel rod perforation in a high temperature environment. This result is taken from Supplement II to WASH-1238. According to Table 6.2.6 of DOE/ET-0028, the availability fraction is divided in half between ^{134}Cs and ^{137}Cs . Table 3.3.8 of DOE/ET-0028 shows a cask inventory of 1.7×10^5 curies and 9.4×10^4 curies per MTHM for ^{134}Cs and ^{137}Cs , respectively. Since a cask contains 4 MTHM, this results in 6.8×10^5 curies of ^{134}Cs and 3.8×10^5 of ^{137}Cs in a cask. Applying the availability fraction of 3×10^{-4} for each isotope yields 204 curies of ^{134}Cs and 114 curies of ^{137}Cs available for release. Since in accident number 6.2.8, 50% of fuel rods are perforated, this results in 102 curies of ^{134}Cs and 57 curies of ^{137}Cs being released in this accident. Table 4.1.1-12 of DOE/ET-0029 shows 200 curies of ^{134}Cs and 110 curies of ^{137}Cs being released. Perhaps the fact that only 50% of the rods are perforated was not taken into account.

We recommend that the radiation dose to the maximum individual resulting from this accident be reevaluated in light of the above comments. (208-NRC)

Response

This comment raises three main issues. First is that population doses have been omitted from the discussion of severe accidents in the draft Statement. Second, population densities used to evaluate radiological consequences of spent fuel accidents are inappropriate for accidents in urban areas. Last, release fractions used for a severe spent fuel accident are in error. Each of these issues is discussed below.

The format used in the final Statement to discuss consequences of predisposal accidents presents both maximum individual and population doses from umbrella source terms. A definition for umbrella source terms is given in Section 3.2.7 of this report. Transportation accident impacts are described in Section 4.8.1.

The population density used for accident consequence analysis is 130 persons/km^2 and is based on an average of populous U.S. regions. Population doses vary approximately linearly with population density. If higher population densities are assumed, high doses would result. Current interim NRC rules prohibit the transport of spent fuel through highly populated areas.

CONSEQUENCE ANALYSIS

Three sources of error are identified in the discussion of the release fractions. The first is that releases of mixed fission products and actinides have been omitted. WASH-1238 (AEC 1972) states that actinides available for release in the pellet-clad gap would be negligible compared to other nuclides. Mixed fission products were conservatively assumed available for release due to leaching of fuel pellet surfaces. The leaching mechanism could only occur if clad failure occurs before the loss of cask coolant. This is not the case in Accident 6.2.8 (DOE/ET-0028). Actinide and fission product releases from the fuel may be possible following an impact that disrupts the fuel pellets and cladding. Accident 6.2.7 in DOE/ET-0028 discusses this possibility.

Second, this comment states that the amount of ^{85}Kr released is understated by Table 4.1.1-12 of DOE/ET-0029. The stated value in the table is low by 400 Ci of ^{85}Kr . The effect of this error is less than 1% of the doses reported in DOE/ET-0029.

The last part of this comment relates to the release fraction for Cs. The release fraction, 6×10^{-4} of all cesium was obtained from NUREG-0069 (NRC 1976b). Half of the fuel elements are assumed to rupture in the accident. This results in a release fraction for all cesium isotopes of 3×10^{-4} . Multiplying this value by the various isotope inventories yields the releases reported in Table 4.1.1-12 of DOE/ET-0029.

Draft p. N.4

Issue

The consequences presented in p. N.4 for severe accidents are based on the dose received by persons from radionuclides released to the atmosphere. Since severe accidents may cause a reduction in shielding efficiency, doses resulting from radiation emanating directly from the cask should also be evaluated. The description of severe accidents in Table 6.2.6 of DOE/ET-0028 indicates a small opening will exist in the cask.

Accident number 6.2.8 is based on number 6.2.7. Number 6.2.8 assumes that no emergency action is taken to cool the cask involved in the 6.2.7 accident. This results in 50% of the fuel rods being perforated in number 6.2.8 instead of only 1% being perforated in number 6.2.7, in addition to 100% of the coolant being released in both accidents. Thus, release fractions in number 6.2.8 should be 50 times higher than in number 6.2.7. Indeed, for ^{85}Kr , ^{129}I , and ^3H the release fractions given are 50 times higher for number 6.2.8 than for 6.2.7. However, although mixed fission products and actinides are reportedly released in number 6.2.7, only ^{134}Cs and ^{137}Cs are reported as being released in number 6.2.8. This can also be seen in Tables 4.1.1-10 and 4.1.1-12 of DOE/ET-0029 which gives the actual number of curies released. Note that Table 4.1.1-10 gives the curies released for accident number 6.2.6, a moderate accident in which only 5% of the cavity coolant is released and only 0.25% of the fuel rods exhibit cladding failure. The table shows fission products such as ^{90}Sr and ^{95}Nb and the actinides such as ^{239}Pu and ^{242}Cm being released. Table 4.1.1-12, which lists the radionuclides released for accident number 6.2.8, the severe accident does not contain any of the

CONSEQUENCE ANALYSIS

additional fission products or actinides listed for the less severe accident. Is this simply an oversight or is the contribution to the dose from these nuclides negligible compared to the dose resulting from the nuclides that are listed?

A study conducted by Battelle's Pacific Northwest Laboratory, "An Assessment of the Risk of Transporting Spent Nuclear Fuel by Truck," PNL-2588, uses release fractions for actinides and fission products other than gases that are significantly higher than those derived from the accidents described in DOE/ET-0028. As previously shown, the release fractions for actinides and mixed fission products in accident number 6.2.8 should be 50-times higher than those used in accident number 6.2.7. Table 6.2.6. of DOE/ET-0028 shows a release fraction of 1×10^{-8} for actinides and mixed fission products for accident number 6.2.7. The release fraction for accident scenarios that involve creep rupture of fuel rod cladding. Since in accident number 6.2.8 it is assumed that 50% of the rods fail, the release fraction for actinides and other fission products should be 1×10^{-5} if the results of PNL-2588 are used. Recommend the basis for the release fraction for actinides and other fission products should be 1×10^{-5} if the results of PNL-2588 are used. The basis for the release fractions used in the PNL study should be resolved.

The following discussion shows that the amount of ^{85}Kr released for accident number 6.2.8, the most severe accident, has been underestimated. Table 6.2.7 of DOE/ET-0028 indicates that 30% of the ^{85}Kr will exist in fuel rod void spaces. Accident number 6.2.8 assumes that 50% of the fuel rods are perforated so that the release fraction reported in Table 6.2.6 of DOE/ET-0028 is 0.15. This table also indicates that the cask inventory given in Table 3.3.8 of DOE/ET-0028 should be used for determining the actual number of curies released. Table 3.3.8 indicates 9.5×10^3 curies per MTHM. Since Table 6.2.6 indicates that a cask will contain a 4 MTHM, this means a total inventory of 38×10^3 curies of ^{85}Kr . With a release fraction of 0.15, this results in 5.7×10^3 curies of ^{85}Kr being released. Table 4.1.1-12 of DOE/ET-0029 shows only 5.3×10^3 curies of ^{85}Kr being released. (208-NRC)

Response

This comment discusses potential reductions in shielding efficiency for a cask involved in a severe accident, compares releases from Accident 6.2.7 to those from 6.2.8, compares release fractions used in DOE/ET-0028 to those in currently available literature and points out an error in the calculation of ^{85}Kr releases for Accident 6.2.8. The first three areas are discussed below. Please see the response immediately above for a discussion of ^{85}Kr releases.

Accident 6.2.8 in DOE/ET-0028 is a cask loss of cooling accident postulated to occur following a moderate impact or derailment that disables the mechanical cooling system. An impact of this type would not be severe enough to significantly reduce the cask shielding and therefore was not considered in Appendix N of the draft Statement. Some reduction in shielding may be possible in a severe impact like that postulated in Accident 6.2.7 but

CONSEQUENCE ANALYSIS

would depend on specific cask construction and behavior. Surface dose rates for casks are designed to not exceed 1000 mr/hr following Type B packaging qualification tests described in 49 CFR 173.398.

The assumption made by the commenter when comparing releases from Accidents 6.2.8 and 6.2.7 is that they are related. However, Accident 6.2.8 is based on Accident 6.2.6. The mixed fission product and actinide releases discussed in 6.2.7 are due to impact damage to the fuel. A severe impact was not postulated for either Accident 6.2.6 or 6.2.8.

Table 4.1.1-10 in DOE/ET-0029 does consider releases of mixed fission products with the cask coolant for Accident 6.2.6. The inclusion of actinides in this table is in error. Excluding plutonium and curium would reduce the dose expected for this accident. Revised Tables 4.1.2-7 and 4.1.1-11 from DOE/ET-0029 are included here. The mixed fission product contribution to the dose from Accident 6.2.8 was negligible.

Estimates of particulate releases from a spent fuel cask involved in a severe accident are different for DOE/ET-0028 and PNL-2588 (Elder et. al. 1978). DOE/ET-0028 assumed that particulates were only generated in an impact environment and that a small fraction would escape the cask cavity. PNL-2588 assumes that impacts and rapid outgassing of fuel elements during creep rupture can release particulates to the cask cavity. A higher fraction of particulates released to the cask cavity were conservatively assumed to reach the atmosphere in PNL-2588 (Elder et. al. 1978). Differences in actinide and particulate releases in the two documents are due to both the assumption of releases during creep rupture and the fraction of nuclides that escape the cask cavity. Particulates were not the largest dose contributor in PNL-2588 (Elder et. al. 1978).

TABLE 4.1.2-7. One-Year Dose and 70-Year Dose Commitment to the Maximum Individual Resulting from a Moderate Accident During Truck Transport of Spent Fuel

Organ	Dose, rem	
	1-Year	70-Year
Total Body	1.4×10^{-5}	7.6×10^{-5}
Thyroid	2.5×10^{-6}	2.5×10^{-6}
Lung	6.8×10^{-4}	9.4×10^{-4}
Bone	1.1×10^{-4}	3.9×10^{-4} *
Skin	4.5×10^{-6}	4.5×10^{-6}

*Original value was in error in Table 4.1.2-7

CONSEQUENCE ANALYSIS

TABLE 4.1.1-11 One-Year Dose and 70-Year Dose Commitment to the Maximum Individual Resulting from a Moderate Accident During Rail Transport of Spent Fuel

<u>Organ</u>	<u>Dose, rem</u>	
	<u>1-Year</u>	<u>70-Year</u>
Total Body	6.8×10^{-3}	4.0×10^{-2}
Thyroid	1.2×10^{-3}	1.3×10^{-3}
Lung	3.4×10^{-1}	4.9×10^{-1}
Bone	5.4×10^{-2}	1.9×10^{-1}
Skin	2.4×10^{-3}	2.4×10^{-3}

DOE/ET-0028, pp. 7.4.24-25

Issue

What are the risks of escape of radionuclides via the fresh airway as a consequence of a transportation underground? (208-NRC)

Response

The operational risks associated with underground transportation of the waste containers have not yet been analyzed in detail because a site-specific repository design and operations schedule are not yet available. However, for this exposure pathway to be important, significant quantities of waste would have to be either vaporized or crushed into very fine particles. The repository does not contain either a source of intense heat or mechanical energy, and thus this pathway is not likely to be important.

DOE/ET-0028, p. 7.4.30, Table 7.4.11

Issue

No estimate of occupational risk is included in the accident analyses nor is there any discussion of possible impacts on continued repository operation, repository closure or retrieval of waste already emplaced. For example, what will such impacts be for accident 7.5 or 7.6?

For accident 7.6, the safety system is a failsafe wedge-type braking system on the cage. What is the maximum allowable braking distance of the cage for the expected release? (208-NRC)

Response

Estimates of occupational risks from the canister dropped down the mine shaft accidents are included in the final Statement. The doses calculated include exposure to airborne aerosol to a limited number of workers in the area during the accidents and doses incurred during decontamination.

CONSEQUENCE ANALYSISIssue

A question was raised as to whether or not the scenario of a nuclear attack on a geologic repository had been considered. (40)

Another commenter suggested that such a scenario not be analyzed because the effects of the bomb itself would greatly exceed the effects of any release from the repository. (17)

Response

As a previous comment points out, the effects of nuclear war (i.e. release of repository contents) would be bounded by the meteorite strike scenario. Therefore, the effects of weapons strike were not specifically calculated.

Issue

One commenter noted that a more likely initiating event for water intrusion into a repository is exploratory drilling or shaft seal failure. (7)

Another commenter felt that the effect of water intrusion where spent fuel elements are placed in steel canisters be given more extensive consideration. (219)

Response

The faulting and flood accident is a worst case scenario which would bound those suggested by the first commenter. The breach of a spent fuel repository followed by water intrusion is discussed in detail in Section 5.5. For purposes of identifying an upper limit with respect to consequences of such an event, no credit was taken for the presence of steel canisters. Multiple barriers (including steel canisters) have been suggested as being able to assume the lack of release of radionuclides to the repository proper for perhaps 1,000 years. On that basis, the consequences suggested for a breach 1,000 years after closure would apply.

Issue

One of the assumptions that makes mined geologic disposal feasible is that radioactive sources placed in a hydrologic environment with slow-moving groundwater will take long periods of time to be transported to the biosphere. Furthermore, retardation effects will slow down (relative to groundwater velocity) the movement of certain species. This basic characteristic is common to all forms of geologic disposal.

The GEIS and its supporting documents fail to analyze flowpaths other than porous flow through intact media. The possible creation of high velocity flow paths by mining operations or fractures created by the thermomechanical response of the rock mass are not considered. Fracture flow driven by thermal convection deserves more attention than meteorite impact or nuclear war as mechanisms for establishing communication between the repository and the biosphere. (208-NRC)

CONSEQUENCE ANALYSISResponse

Fracture flow surely has an acceptability level lower than porous media flow. Fracture flow velocities will have to be slower in ratio to retardation coefficients of the most dose-significant nuclide. If that nuclide was a nuclide with no retardation, then the fracture flow velocity would have to be slow enough to allow significant decay. Clearly a repository siting and design parameter would be the amount of migration retardation available. The Statement assumes that the repository is placed in a hydrologically unacceptable location. Thermo-mechanical effects are relatively near field.

Issue

Measures of performance used in the GEIS and its supporting documents made it difficult to judge statements that claim "no deleterious effects." For example:

1. Dose received by maximum individual. This seems to be someone using a water supply separated by 10 miles of porous flow from the repository. Note that fracture flow with its lower retardation factor is not considered.
2. Concentration at three miles from boundary. This was used in TM-36, Volume 21. In this case, ⁹⁹Tc occurs near the surface at 400-600 years and exceeds maximum permissible concentrations by one thousand (TM-36/21 pgs. xiv, 8.5-8.6). (208-NRC)

Response

The repository will be sited to reduce the possibility of fracture flow impacts.

Issue

The Statement does not address potential accidents at the spent fuel facility such as zirconium catching fire upon heating up or an airplane crash. (55)

Response

Analysis of events (accidents) at a spent fuel storage facility may be found in DOE/EIS-0015 (DOE 1980b). Accidents are described as either "operating" or "severe." The two scenarios above do not appear under either the operating or severe category.

Issue

The ground-water transport analysis in the main body of the statement uses a path length of only 10 km. The analysis should use a longer flow path for the base cases and discuss consequences of shortening of the path due to tectonic and/or climatic change. (218-DOI)

CONSEQUENCE ANALYSISResponse

As noted the ground water transport analysis used only a 10 km path length. It had been decided that because of timing only one case would be modeled and it was further decided that the shorter length would receive less criticism than an overly long path length. As suggested something of a sensitivity analysis on consequences versus path length would have been useful.

Issue

The worst-case accident scenarios are unrealistic and should be reformulated.

Analyses of worst-case accident scenarios should be carried out on the assumption that application of state of the art techniques will limit dissolution rate of waste forms, including proposed packaging of spent fuel. The availability of a flow of $2.8 \text{ m}^3/\text{second}$ of ground water through any site chosen for a repository is unrealistic. The pressure head to drive such a flow would not be available. Even when underground rock formations become saturated with water, the volume will be inconsequential, and flow rates will be generally close to zero. (219)

Response

It is hoped that a flow rate of $2.8 \text{ m}^3/\text{s}$ (100 cfs) would indeed be unrealistic as a ground water flow rate through any site chosen for a repository. This scenario was developed as an extreme case and while highly improbable is not believed impossible when viewed as an occurrence over millenia, due to tectonic and/or climatic changes.

Issue

Accidents occurring soon after the repository is closed should be given more prominence.

A less dramatic but more likely failure occurring at an earlier time-such as the slower but more extensive flushing of a saturated repository-should be examined with respect to consequences and risk. An early chapter should describe the step-by-step process by which the dose estimates are developed. The model should be illustrated diagrammatically and, wherever possible, relevant source terms and rates should be indicated. (219)

Response

In most instances the consequences for accidents were calculated for a time just after repository closure. A case was examined where a relatively large flow of water penetrated the repository and the consequences presented. It is not believed credible that even a saturated repository would contain significant amounts at an early time.

The size and organization of the EIS precludes putting all material in early chapters; a relatively detailed presentation of dose models was presented in Appendix D.

CONSEQUENCE ANALYSISIssue

One commenter noted that consequences of hypothetical events and resulting absolute doses are presented, but no quantified predictions of the probability of these events happening are given. The U.S. Geological Survey recommendation to provide a candid assessment of the uncertainties associated with the spectrum of alternative out-comes of geologic containment is an objective safety criterion. These probabilities and their limits should be given at 10, 100, 1,000, 10,000, 100,000, and 1,000,000 years for each of the barriers of waste form and container, medium, and site geohydrology, that is, through to the time of radioactive decay of the waste to natural levels of radiation density in the ground. (28)

Response

To the contrary, where available, predictions of probabilities were given in the draft and appear again in the final Statement (see Section 5.5). Some events do not lend themselves to estimates of probability and the associated societal risks remains a more subjective assessment. In a number of instances large errors in probability would not result in a conclusion that a serious societal risk would exist.

Aside from multiplying the number of years times the probability per year little can be done at this stage to provide probabilities over time periods suggested.

Reference to reduction of radiation levels to natural levels in the ground was also made by the commenter. This aspect is treated briefly in Section 3.4. However, the reader is cautioned that while such hazard indices provide perspective, a direct comparison of waste nuclides and naturally occurring nuclides needs qualification.

Issue

The release of ^{222}Rn from soil and/or rock should be evaluated for a leaking repository. The entrance of radon into homes through foundation walls or base slabs may be the most important source of future human exposure. (196)

Response

^{222}Rn is not a significant contributor to human exposure.

Issue

Several commenters stated that the Statement should clearly outline the sensitivity of the parameters used in the calculations and to the extent possible how conservatism offsets uncertainties in the analysis. (38, 43, 58, 97, 113-EPA, 124, 202-HEW)

CONSEQUENCE ANALYSISResponse

The DOE believes that what was done was correct within the limitations of the data. An error of a million or more would not change the conclusion that the accidents analyzed would have an insignificant mathematical expectation of societal risk.

Y/OWI/TM-36/21Issue

The results of simplified calculations given in Y/OWI/TM-36/21 show ^{99}Tc exceeding acceptable concentrations 3 miles from the center of the repository 400-600 years after recharge. Top quote from page 8-5: " ^{99}Tc , due to its long half life and unity retardation coefficient exists in all layers of the generic stratigraphic columns studies (shale, granite and basalt) at concentrations near or equal to the source activities. The maximum source activity for ^{99}Tc used in this study is approximately 0.2-0.3 Ci/ml (Section 7.0) which is at least 10^3 times greater than an acceptable level. The first arrival of ^{99}Tc occurs in the near surface layers between 400-600 years after repository decommissioning and resaturation and at concentrations near or equal to that of the repository source activity." This would appear to indicate unacceptable repository performance. An explanation should be given of how this will be remedied or why this analysis is not believed to indicate a problem. (208-NRC)

Response

Unacceptable repository performance would result only under a very improbable set of events. The scenario is indicative of a faulting and groundwater intrusion accident rather than expected parameters.

DOSE CALCULATIONSDraft p. 1.11Issue

The annual dose from naturally occurring sources is given as 100 millirem/yr. It should be pointed out that this is a lower limit estimate and not an estimate of typical exposure. (166)

Response

Natural background radiation was used to provide a perspective. According to NCRP-45 (NCRP 1975) 100 millirem/yr would over estimate rather than under estimate the typical exposure.

Draft p. 1.11Issue

It is stated that 0.1 rem per year will be used as the background dose rate. Over 70 years this will result in an exposure of 7.0 rem. One percent of this exposure is 0.07 rem. The 0.1 rem the maximum individual receives as a result of transportation is greater than 1% of background exposure, not less than 1% as stated in the Statement. (208-NRC)

Response

The comment is correct. $\frac{0.1 \text{ rem}}{0.1 \text{ rem} \times 70} = \frac{0.1}{7.0} = 1.4\%$

Statements comparing 0.1 rem to 70 years background now indicate that the dose is approximately 1% of background.

Draft p. 1.11Issue

In comparing natural and manmade doses, person-rem is the sum of doses to individuals in the population and so is a function of both individual doses and population size. The extra 260,000 person-rem in Colorado compared to Louisiana is meaningless in that population size selection was arbitrary. Why not use New York and Hawaii? (113-EPA)

Response

Comparisons of population doses caused by naturally occurring sources in various regions of the United States were deleted.

DOSE CALCULATIONSIssue

Several commenters questioned comparisons of manmade and natural doses.

Draft p. 1.11--The comparison of natural radioactivity received in any particular sector with that artificially induced is totally fallacious and has no place in the serious consideration of radiation doses. (144)

Draft p. 1.11--All values in this section should be based on the same units. The term man-rem/yr/individual may be confused with population dose. (196)

The frequent comparisons with background radiation are irrelevant. (8)

When comparing resulting doses to the natural background, the concentrations should be compared to the maximum permissible concentrations given by the International Commission on Radiological Protection (ICRP). (28)

Response

The comparison of dose from natural sources to that received from waste management operation is provided to give the reader perspective. Such comparisons are not intended to imply that background radiation levels are harmless and were made when doses were large enough to warrant comparison.

The use of man-rem/yr/individual was unfortunate and was avoided in the final Statement.

Comparison of MPC's must be done nuclide by nuclide to be meaningful. Comparison of doses obviates such long lists which would not add clarity to the presentation.

Draft p. 1.11Issue

Having set an (arbitrary?) annual dose radioactivity at 0.1 man-rem/yr/individual, a new unit of man-rem/yr is referred to in sentence 4, followed by another unit, man-rem/yr population. Will people in Colorado receive natural radiation at the dose rate of 258,000 man-rem/yr, with an additional burden of 2,000 man-rem/yr from reprocessing? Will Louisianans receive natural radiation at the rate of 0.1 man-rem/yr/individual with an additional 2,000 man-rem/yr population (what population?) from reprocessing? The sentence begs for more information: How many individuals in Colorado and Louisiana? How many individuals in 'population?' It seems all apples and oranges and will either totally confuse the general reader or provoke rash quoting of the raw numbers on a comparative basis when there is no basis for comparison. (181)

Response

The material comparing population dose from natural background between states has been removed from the final Statement. The thrust of the regional material was to indicate that

DOSE CALCULATIONS

people (as a group) could reduce their collective dose by moving from Colorado where the collective dose rate was given as 650,000 man-rem/yr to Louisiana where the collective dose rate was given as 380,000 man-rem/yr but apparently do not choose to do so. This was removed because it is believed that most of the population does not have the knowledge upon which to base such a judgement and therefore the illustration was misleading.

Where a reprocessing plant contributed 2,000 man-rem/yr to a regional population the total for Colorado if the dose rate from background was 258,000 man-rem/yr would be 260,000 man-rem/yr.

Draft p. 1.19Issue

Some of the main conclusions given in the summary concerning radiological impacts are not readily traced back to the supporting text, e.g., p. 1.19, lines 14 to 17. The text in the summary section (Section 1.3) states that, "Calculated radiation dose to the total population from routine operations including transportation, assuming that all facilities are located in the same region (a highly conservative and unlikely scenario) amount to no more than about 0.3% of the dose factor of less than 15 among fuel cycle options." Although the summary gives no reference to where the supporting text for this conclusion is, it appears that the supporting data base is in Tables 3.1.87 (summarizing environmental effects from routine operations). However, several entries (e.g., see U and Pu recycle column on p. 3.1.215) give regional population doses (6×10^{-4} man-rem) that are greater than 0.3% of background as quoted above. (208-NRC)

Response

The values listed in draft Tables 3.1.84 through 3.1.87 are rounded to one significant figure. This rounding process caused changes in the listed numbers which affected the changes noted. Therefore, comparison of the ratio of regional dose to natural background from these numbers could easily be high by the amount found in the stated example.

Draft p. 1.19Issue

The commenter noted that a statement is made regarding the occupational population dose and resultant health effects without identifying the population base. (34)

Response

The work force among which the population dose and health effects were calculated amounted to about 18,000.

DOSE CALCULATIONSDraft p. 1.20Issue

Some of the numerical values on page 1.20 (e.g., maximum individual dose) cannot be traced to Section 3.1.5. (208-NRC)

Response

The numerical values on page 1.20 of the draft Statement are repeated in draft Table 3.1.54. These tables give only worst-case results. For the maximum individual the dose presented is the dose during the first year of exposure. Values for faulting and flooding and drilling were taken from draft Tables 3.1.41 and 3.1.51, respectively. The first-year dose from solution mining is not given elsewhere.

Draft p. 1.20Issue

In Item 3 of Table 1.4, the regional natural radiation dose is calculated for three generations. In Item 2, doses are calculated for only one generation (70-yr total body) resulting in inequitable bases for comparison. (208-NRC)

Response

In draft Table 1.4, page 1.20, regional natural radiation dose under the columns labeled meteorite impact, fault fracture and flooding, and drilling accident should be 1.4×10^3 to 1.1×10^4 .

Issue

Several commenters questioned the regional population assumption.

Draft p. 1.20--The regional population assumption appears to be far too conservatively high. (147)

Draft p. 3.1.100--The use of a reference population of 2,000,000 people within 80 km of a repository to calculate doses is questionable when nuclear plants in the Northeast U.S. have a surrounding population (within 50 km) of 50,000,000 people. (30)

Response

The reference environment of 2,000,000 was taken from actual data (adjusted slightly for location) at a mid-west reactor site.

DOSE CALCULATIONSDraft p 3.1.100Issue

The regional population for the types of releases considered most likely in the long term in waste disposal is the population in a river basin rather than the population within an 80 kilometer radius of the plant. (113-EPA)

Response

The DOE agrees with the comment. However, in the reference environment the majority of the population lives along River R, downstream of the release point.

Draft p. 3.1.101Issue

Why is bone an organ of principle interest? According to the BEIR work, more health effects would be expected from a dose to red marrow than from the same dose to bone. The liver should also probably be considered a significant organ. (113-EPA)

Response

The bone dose is used because it supplies an adequate estimate of doses to various portions of bone. Liver dose is not significant except for a few actinides.

Draft p. 3.1.120Issue

The draft states that the regional population dose for a geological repository during construction and operation is 100 man-rem. However, no reference is given to the basis for this estimate. For example, how much radon is estimated to be released during construction and operation at the repository? (208-NRC)

Response

The data requested is contained in DOE/ET-0029, Sections 4.4 and 9.0.

Draft pp. 3.1.136-172Issue

The radiological impacts should be based on the worldwide population and over all time following release. (8)

DOSE CALCULATIONSResponse

In preparing the final Statement, the feasibility of expanding population and time parameters in dose calculations was examined and found to have no appreciable effect on the conclusions.

Draft pp. 3.1.142-144

Issue

There seems to be inconsistency in the values given in Tables 3.1.30 to 3.1.32. Comparing Tables 3.1.30 and 3.1.31 for 100, 100,000, and one million years the 70-yr doses are just about 70 times the first year doses. At 2050, however, they are 250 to 750 times the first-year doses. Comparing Tables 3.1.31 and 3.1.32, in order to obtain the regional population doses in Table 3.1.32 from the maximum-individual doses given in Table 3.1.31 at the year 1050 it requires from 10 to 27 persons, which simply is not reasonable. The 1,000 year values require from 40,000 to 60,000 persons--much more reasonable. At 100,000 years the values range from 600 to 5,000--much too much variation and too low; at one million years the range is 500 to 17,000 persons, again too large a range and too low. (154)

Response

Draft Tables 3.1.30 to 3.1.32 were checked for errors and consistency. The significant pathway to man changes with time and as a consequence ratios between maximum-individual dose and per-capita dose are not constant.

Draft p. 3.1.150

Issue

Doses to a maximum individual are not the best measure of the impact of this accident. Total population dose and integrated population dose are most significant. The emergency dose limits of 100 rem and 25 rem apply implicitly to the case where only one or a few people are exposed. (113-EPA)

Response

Dose to the maximum individual is one measure of impact, the population dose another; both were presented. Reference to emergency dose limits was provided for perspective only to the maximum individual.

DOSE CALCULATIONSDraft p. 3.1.162Issue

One commenter questioned the total body dose (in Table 3.1.47) at one million years exceeding the dose received at one hundred thousand years. (43)

Response

After extremely long periods of disposal, the decay to ^{226}Ra of ^{238}U plus selective transport of nuclides could result in a higher dose after one million years than after one hundred thousand years.

Draft p. 3.1.163Issue

The area used for regional dose estimates is too small. (8)

Response

The 50-mile radius has been standard for some time. The Statement did, in some instances, examine the dose to the eastern U.S. population and the world wide population from gaseous releases.

Draft p. 3.1.163Issue

It is not proper, to assume doses to the regional population from a ratio basis with total body dose. (113-EPA)

Response

DOE would suggest that the method used is imprecise and may have broad uncertainty bands. However, DOE believes the results are illustrative and perhaps conservative.

Draft p. 3.1.164Issue

Diverting the entering stream until it is diluted by another stream does not change the population dose. It merely means smaller doses to more people. (113-EPA)

DOSE CALCULATIONSResponse

Dilution will likely reduce concentration of radionuclides in the water by adsorption or other mechanisms while the total water usage will remain the same. Therefore, total radionuclide intake is expected to be less as in population and individual dose. If it were simply a matter of dilution in water, the comment would indeed be correct.

Draft pp. 3.1.165-168

Issue

Doses to individuals are said to be due in the first year predominantly to direct radiation. Food pathway doses to bone "increase substantially" during the 70-yr dose accumulation pathway. It is not clear why. We have been able to roughly check these values. (154)

Response

The increase in 70-yr dose commitment to bone is because of the buildup of actinides in the bone. No clearance of actinides from bone is assumed by the models used.

Draft p. 3.1.179

Issue

Since the releases in Table 3.1.58 are suspect, so are the population dose values in Table 3.1.59. (154)

Response

No problem was found with draft Table 3.1.58 and the dose values in draft Table 3.1.59 are believed to be correct.

Draft p. 3.1.192

Issue

The overstated value for ^{14}C in Table 3.1.68 would imply that Table 3.1.69 should be in error, particularly for world-wide dose which is totally dominated by ^{14}C , but the quoted figures cross check with Section 5.4.2.4 of DOE/ET-0029. Before issuing the final EIS it should be determined which figures are correct. (154)

Response

The value for ^{14}C was given incorrectly and has been adjusted. However, the world-wide dose in draft Table 3.1.69 is correct as shown in the draft.

DOSE CALCULATIONSDraft p. 3.1.192Issue

The most severe accident postulated for predisposal operations is a transport accident involving HLW. Maximum-individual 70-yr dose is 7 rem. This is significantly lower than normally seen for this accident. Nor does it seem reasonable in relation to the spent fuel transport accident discussed on pp. 3.1.178-179. (154)

Response

The dose to the maximum individual from a severe transportation accident involving HLW is 37 rem. This typographical error was corrected for the final Statement.

Draft Appendix DIssue

While the calculation models employed may be adequate, in light of the uncertainties inherent in the input data, they are not state-of-the-art, as claimed. For example, the calculation of the 5-cm gamma dose as the total body dose for air immersion could be improved by the use of an existing code employing the TGLD model. It does not explicitly treat the daughter products formed after inhalation as do more complete codes. (113-EPA)

Response

Use of programs giving specific organ doses from external radiation do not give significantly different results than the 5-cm estimate to warrant their use (see NCRP-45 pp. 108 and 109, NCRP 1975 in reference list). The DACRIN code dose considers daughter product radiations formed in organs of interest. Daughters produced in transit to the organs are not considered, but for the majority of cases this omission is not significant.

To avoid unnecessary argument, the expression "state-of-the-art" in reference to the codes used has been dropped.

Draft Appendices D and FIssue

Although the sections on radiological models indicate that all pathways were considered, the contribution of various pathways to the total dose is not given in the document. Additional information on the radiological analysis for scenarios (e.g., source terms, concentrations of nuclides for different locations, solubility classifications of particulates, etc.) would help document the major conclusions concerning radiological impacts. (208-NRC)

DOSE CALCULATIONSResponse

The contribution of various pathways to total dose is dependent on the radionuclides released and varies markedly among the many release situations considered. Inclusion of this information for all dose tables would lead to excessive bulk and is not practical.

Draft p. D.4Issue

On page D.4, the draft Statement states that the methodology used to calculate the direct radiation dose to persons along the shipping route follows that developed in WASH-1238. Subsequent to the issuance of WASH-1238, an environmental statement on transportation of radioactive material has been published by the NRC. This statement is NUREG-0170, "Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes." The latter document uses a more realistic demographic model in determining population density along the transport route and an improved method for evaluating integrals used in the model. We recommend this refined methodology be used in the GEIS in assessing the radiological impact to the population residing near the transport route. (208-NRC)

Response

The DOE disagrees. While the demographic model used in NUREG-0170 may be more realistic and the method of evaluating integrals improved, the results are similar.

Draft p. D.4Issue

One commenter raised a question as to the accuracy required for input parameters (to dose models) derived from environmental measurements of radioactivity. (43)

Response

The most accurate data that is available is required and even that results in uncertainties of perhaps overestimation by a factor of 100 or underestimation by a factor of 10.

Draft p. D.5Issue

Population density assumptions used in determining the radiological consequence of transporting radioactive wastes are given on page D.5. A value of 330 persons per square mile is used for the Eastern U.S. and California and 110 persons per square mile is used for the Western U.S. The environmental aspects presented in DOE/ET-0029, page 4.1.7, use a

DOSE CALCULATIONS

population density of 90 person per square km (230 persons per square mile). If this is a weighted average, the weighting factors should be given so that their validity can be evaluated. (208-NRC)

Response

The population density, 90 persons/km², is a weighted average. Fifty percent of the AFRs were assumed to be in the West (40/km²) and 50% were assumed to be in the East (130/km²) for an average of about 90/km². These are equivalent to the 330/mi² and 110/mi² given on draft p. D.5.

Draft p. D.8Issue

The model used to estimate the population dose commitment from ¹⁴C is too conservative (i.e., overestimates the impact). If dilution by the Suess effect is not considered and the total number of health effects is integrated over all time, the release of 1.4 MCi (from Table 3.1.68) would result in about 5×10^6 deaths, assuming a stable world population of 6.4×10^9 people. A more realistic comparison might be to the natural production of ¹⁴C and associated health risk. (113-EPA)

Response

Suess effect is dilution of ¹⁴C with increased CO₂ from fossil fuels. This could dilute ¹⁴C by a factor of 0.5 - 1.0. Certainly this small change is far less conservative than integrating population dose out to time infinity like EPA suggests. Even 5×10^6 deaths are small by their method. For example, a constant population implies death rate = birth rate = 1.9% per year, or $(1.9 \times 10^{-2}/\text{yr}) \times (6.4 \times 10^9) = 120$ million deaths each year. This is 24 times the 5×10^6 value of EPA in the first year alone. The model use was designed for reasonable times not infinite times, and the DOE feels that it works well for the times for which it was used. If health effects were integrated to infinite time, a model which took into account the sinks for ¹⁴C (such as the deep ocean), which remove it from the human biosphere, would have been used.

Draft Appendix IIssue

This appendix as well as Appendix G to DOE/ET-0029 present impacts at time periods of 10² years, 10⁵ years, and 10⁶ years. Sometimes 10⁴ years is discussed. Since preliminary versions of the EPA standard for high-level waste specifically reference the 10⁴-yr period, presentation of cumulative dose calculations for this time period for all cases studied would be prudent. (208-NRC)

DOSE CALCULATIONSResponse

The time required to calculate doses for 10^4 years after production of the draft was not available. It is suggested that interpolations between results at 1000 and 100,000 may be sufficient for present purposes.

Draft p. N.4Issue

Repeated reference to discussion of trucker's dose on p. N.4 is misleading. The reference on p. 13 indicates the discussion on p. N.4 explains the overestimate of the dose and the reference on p. N.16 indicates the discussion on p. N.4 is based on experience. The actual discussion on p. N.4 satisfies neither of these descriptions. (208-NRC)

Response

All of the truck drivers' doses in DOE/ET-0029 and the Statement are based on the conservative WASH-1238 (AEC 1972) overestimate. The discussion on draft p. N.4 is an attempt to point out that the model results in an over estimate and actual values are smaller.

The following quote from WASH-1238 (AEC 1972) is the basis for this statement.

"Experience in truck transport of radioactive materials has shown actual doses to drivers to be 0.1 of the regulatory limit. Use of the regulatory limit of 2 rem/hour in any normally occupied space is therefore considered conservative."

Draft p. N.4Issue

Did this result take into account the growth of population along the transport route during the 70-yr period? (208-NRC)

Response

No, it did not.

Draft p. N.4Issue

An inconsistency exists between these two paragraphs. In the third paragraph, the draft states that population doses are calculated based on the permissible limit of radiation. Individual doses given in the fourth paragraph are taken from WASH-1238 which used dose rate values derived from experience rather than permissible limits. In addition, these WASH-1238 numbers were obtained from considering average exposures resulting from the transport of fuel and waste from power reactor. Since the discussion on page N.4 of the GEIS

DOSE CALCULATIONS

concerns transportation of spent fuel, it would be better to examine the WASH-1238 analysis of exposures caused by transport of spent fuel which can be found on pages 40-42. (208-NRC)

Response

The intent is to illustrate that the approach used was conservative and to illustrate the magnitude of doses incurred.

Draft p. N.9Issue

Last paragraph: The accident postulated here results in 37 rem to the total body. Table 3.1.224, shows the results of the worst-design basis accident, which for SHLW--severe impact and fire is 7 rem. (208-NRC)

Response

This is a typographical error in draft Table 3.1.88. The dose should be 37. See Table 7.1.1-5 in DOE/ET-0029.

Draft pp. N.13, 16, 21Issue

On pages N.13, N.16 and N.21, reference is made to page N.4 and a discussion on dose to truckers. The reference on page N.13 indicated the discussion on page N.4 explains the overestimate of the dose and the reference on page N.16 indicates the dose discussed on page N.4 are based on experience. These references are misleading since the discussion on page N.4 satisfies neither of these descriptions. Is the reference intended to apply to WASH-1238 which is the basis for the truckers dose given on page N.4? (208-NRC)

Response

Yes. See the response to comment above referring to p. N.4.

Draft p. N.13Issue

"Doses to the maximum individual...and population dose are comparable." This statement does not make sense, since there is a 10,000 times difference between the maximum individual dose and population dose in Table N.12. This should be clarified in the final EIS. (113-EPA)

DOSE CALCULATIONSResponse

The statement should read, "The dose to the maximum individual for rail and truck shipments is comparable as are the population doses attributed to each transport mode."

Issue

The several appendices which support the long-term impact assessment need to be coordinated so that their results are directly comparable. Some cumulative doses are for 50 years, some for 70 years. Different times are referenced. The total picture is confusing and leaves many questions about the internal consistency of the supporting calculations. (208-NRC)

Response

The problem is due to computer models availability. Differences in doses caused by the parameters used in the computer models are discussed in the final Statement.

Issue

One commenter stated that background radiation and doses from waste disposal analyses are in different units throughout the report. (13)

Response

It is believed that the analysis is best served by using different doses for different purposes. Accident doses are often given in rem/yr to the maximum individual with emphasis on the first-year dose where it is believed that such a dose could have acute effects. Population doses are expressed in man-rem, being the sum of the number of people receiving a given dose over the population of interest. The use of natural background is to provide perspective to the doses tabulated.

Issue

There are substantial problems in the calculation of radiation doses and health effects to the public. The time-integrated population dose is frequently neglected. Furthermore, population doses are not always expressed as fatal, non-fatal, and genetic health effects; we think that they should be.

The Draft EIS appears to indicate that the major hazards occur in the first few hundred years while ^{90}Sr and ^{137}Cs are present. As a result, long-lived nuclides, such as ^{99}Tc and ^{129}I , are neglected despite the fact that they can be geochemically mobile under some circumstances. (113-EPA)

DOSE CALCULATIONSResponse

There are no substantial problems in the calculation of radiation dose and health effects to the public as EPA has stated. Most of the general comments presented are answered in response to specific EPA comments. In general, DOE does not feel that emphasis on time-integrated population doses is required. In some cases doses to first, second, and third generations are given (70 yrs/generation) and in a few cases doses were integrated over 10,000 years.

Doses caused by ^{99}Tc and ^{129}I were not neglected. These were included in the source terms, but if they did not contribute at least 1% to doses calculated they were not listed in the source terms presented in the Statement.

Issue

A commenter suggested that a presentation of radiation dose summaries in a separate section of the document could provide the public with a better understanding of the health impact from various operations. (202-HEW)

Response

For the disposal technologies analyzed, the Statement presents the radiological impacts under the sections titled "Environmental Impact of Construction and Operation," (5.4, 6.1.1.4, 6.1.2.4 - 6.1.8.4) and "Environmental Impacts Over Long Term" (5.5, 6.1.1.5, 6.1.2.5 - 6.1.8.5). The radiological impacts of predisposal activities are addressed in Sections 4.7 and 4.8. Section 7.4 outlines the radiological impact for entire waste management systems.

RISK PERSPECTIVESIssue

Several commenters noted that the period of time cited as critical for containment of radioactive wastes within a repository was not presented in a consistent fashion.

Draft p. 1.5--Five hundred years rather than 1000 years in the period that is most critical. On p. 1.9 it is stated that high level wastes must be retained safely for up to one million years. This is patently ridiculous. (147)

Draft p. 1.5--The 1000 years noted as the time period necessary for decay of ^{90}Sr and ^{137}Cs is too long a period. (154)

Draft p. 1.9--The time required for isolation is inconsistent with the discussion presented in Section 3.1.3. (124)

Draft p. 1.9--Uranium, plutonium, and the balance of the TRU's do not need to be isolated for up to one million years. (154)

Draft p. 3.1.16--The statement that "High level wastes must be kept isolated from the biosphere for a long time period, perhaps hundreds of thousands of years" is at variance with statements made elsewhere in the draft document (eg., p. 1.9-- up to one million years and p. 1.28--500 years). (198, 218-DOI)

Draft p. 3.1.64--It is stated that "...after several hundred years of decay, the wastes do not exceed the natural radioactivity of the ones from which they came." Elsewhere in the report, however, references is made to isolation times of one million years or isolation until the waste has decayed to harmless levels. (38, 218-DOI)

Draft p. 3.1.65--The statement was made that "containment times of 500 years are the most important." However, on page 3.1.59 it is stated that a significant release "could occur at 1000 years" and on page 3.1.64, it stated that after "700 years, the radioactivity in the repository poses a greatly reduced threat." Some consistency should exist in the document for the period of concern and basis for arriving at this time should be clearly delineated. (208-NRC)

Statements which point out the essentially "back to nature" radioactive decay time of 500 years may not be clear to the general public. (166)

Response

The material has been thoroughly revised and attention paid to such consistency. The "500 year" and "700 year" numbers are from the referenced work of other authors. The value used in this Statement is 1,000 years.

The 1,000 years allocated for the decay of mixed fission products typified by ^{90}Sr and ^{137}Cs is based in part on conservative calculations of the percentage impact of these nuclides on the total base and on proposed canister design criteria. The language used on draft p. 1.5 has been modified to omit reference to general acceptance. For the inventories involved, 500 years would be adequate (or more than) for just ^{90}Sr and ^{137}Cs .

RISK PERSPECTIVES

Repositories are designed with the expectation that radionuclides will not reenter the biosphere. Reference to "... isolate ... for several millenia ...one millenia" is superfluous and has been removed.

Draft p. 1.16

Issue

The hazard indices discussed in Section 3.1.3.4 and mentioned here are at best crude estimates. The hazard of a material is based on three factors: 1) the quantity of the materials available, 2) the toxicity of the materials, and 3) the pathways between the materials and human beings. Hazard indices which do not consider all of these (and it is difficult to think of a generic hazard index which would be useful for specific pathways) are not particularly useful. (113-EPA)

Response

While the hazard indices used are not refined estimates, they do provide some insights to the hazards of the wastes.

Draft p. 1.16

Issue

The following statement is made--"The conclusion is that the available lethal doses in radioactive waste are far less than the available lethal doses in toxic nonradioactive chemicals now being handled routinely by society as shown in draft Table 1.3. Further, radioactive wastes decay with time whereas toxic chemicals have no half-lives and hence their quantities remain unchanged with time."

- Is the value in Table 1.3 for radioactive waste based on deaths caused by the radiotoxicity or the chemical toxicity?
- How does this value behave with time?
- Provide references for Table 1.3.
- Available lethal dose is defined as (the number of) potential deaths if dose is uniformly administered.
 - What does this mean?
 - What "dose" is uniformly administered?
 - Administered to what populations?
- How many available lethal doses result from the eventual stable daughter products of the radioactive waste? (208-NRC)

RISK PERSPECTIVESResponse

- The deaths postulated are due to radiotoxicity.
- The value decreases with time.
- References can be found on p. 3.1.79 of the draft Statement.
- The available lethal dose is based on the total inventory of a toxic substance available divided by the amount of that substance known to cause death in man. This gives the number of available lethal doses.
- One could determine amount of stable daughter products from radioactive nuclides in a repository and compute the available number of lethal doses. These would be insignificant when compared to magnitudes of other numbers in the table.

Draft p. 1.16, Table 1.3Issue

One commenter did not understand how the annual available lethal doses from a "hypothetical all-nuclear electric economy" could be as low as the 10^7 figure cited. (211)

Response

This table was in error. The 10^7 figure should be 10^{10} lethal doses.

Draft p. 1.16Issue

Is the statement--"available lethal doses in radioactive waste are far less than the available lethal doses in nonradioactive chemicals now being handled"--referring to chemical or radiological toxicity? (197)

Response

As noted above radiotoxicity is being addressed.

Issue

Many letters commented on the comparisons presented between radiological wastes and hazards from toxic wastes, as well as lives lost in automobile accidents.

Draft p. 1.16, Table 1.3--The authors have performed a service in putting the hazards of radioactive waste in perspective with the much greater hazards from other forms of toxic waste. It was also entirely appropriate for the GEIS to point out that, "radioactive wastes decay with time whereas toxic chemicals have no half-lives and hence their quantities remain unchanged with time." (198)

RISK PERSPECTIVES

Draft p. 3.1.41, 64-74--We note with considerable interest the following: Each year 55,000 persons lose their lives on highways. Although tragic to families, this does "not represent a significant societal loss to a population of 200,000,000." On conservative bases and even at relatively imprecise levels of knowledge, the annual loss of life from a waste repository will be "a small fraction of one." (emphasis added). We agree completely with this analysis. But one must then ask why waste disposal has not been put in proper perspective. The risk associated with the disposal of nuclear waste has been exaggerated out of all proportion-by many orders of magnitude. DOE should come right out and state this obvious fact. (154)

Draft p. 3.1.65--The relative toxicity of plutonium and lead should be more widely promulgated. Although it does not mean that concern for plutonium be reduced, it does put the problem in better perspective. (32)

The statements which point out the advantage radioactive wastes have over toxic chemicals should be emphasized. (166)

Draft p. 1.16, Table 1.3--This comparison is superfluous and misleading. (30)

Draft p. 1.16, Table 1.3--Comparison of lethal doses from a postulated all nuclear economy is absurd. This number should be reduced. (147)

Draft pp. 1.16--There appears inappropriate comparisons of the lethal doses from a variety of toxic materials and lives lost in auto accidents with deaths caused by nuclear waste disposal. The real question is whether a considerable area of water shed will be contaminated. (40)

Draft p. 1.16--The fact that we run a greater risk of lethal contamination from the environmental presence of arsenic or cyanide than from high-level radioactive wastes will not assuage the public's concern over the safe disposal of these wastes. (41, 170)

Draft p. 1.16--Your figures are misleading because some of the chemicals cited are not lethal if diluted; others are not lethal when in certain chemical compounds. (128)

Draft p. 1.16--The statement is made that--"The conclusion is that the available lethal doses in radioactive wastes are far less than the available lethal doses in toxic nonradioactive chemicals now being handled routinely by society as shown in Table 1.3." Chlorine, phosgene, and ammonia are unstable in contact with the atmosphere and consequently do not persist in their uncombined state. Inclusion of these substances in that table negates its credibility. Radioactivity, of all types and in all forms, leaves a persistent trace in cell structures and its ambient strength is eternal relative to the human span of life. The listed chemicals combine with other elements and become harmless. (144)

Draft p. 1.16--The statement--"Radioactive wastes decay with time whereas toxic chemicals have no half-lives and hence their quantities remain unchanged with time."--seems also to be a distortion of the nuclear question in that it equates chemical toxicity with radioactivity. As I understand it, they are two separate phenomena. (145)

RISK PERSPECTIVES

Draft p. 1.16--The results shown in Table 1.3 are interesting but can be misleading. Several of the substances listed are innocuous when their concentrations in the environment are sufficiently low, as for that matter is nuclear waste. We do agree with the premise that many other products and wastes are as significant an environmental problem as that associated with high level waste. However, Table 1.3 does not support this conclusion in a meaningful way and reliance on this simplistic comparison may be counter-productive. (154)

Draft p. 1.16--The statement--"Radioactive wastes decay with time whereas toxic chemicals have no half-lives and hence their quantities remain unchanged with time."--does not recognize the fact that many chemicals (toxic or not) undergo transformations into other chemicals. (211)

Draft p. 1.16, Table 1.3--This table could be misleading, in the sense of comparing apples to oranges. Chlorine gas, for instance, will rapidly deteriorate in most environments because of its high reactivity. Phosgene and ammonia are also non-persistent. Most of these substances can easily be treated to render them relatively harmless. (218-DOI)

Draft p. 3.1.41--Comparison of consequences from radiation in comparison with those from automobile accidents is invalid. There are two primary aspects to the establishment of bases (for determining acceptable consequences for operational and post-operational radiological accidents):

1. How much will society accept on an absolute basis?
2. How much better than this can the technology provide? (113-EPA)

Draft p. 3.1.41--It is recommended that "bullet" three be removed, since these are major NEPA considerations, not just information to be discussed under the subject of mined geologic disposal. (124)

The toxicity comparison on p. 3.1.65 is interesting, but rather irrelevant. (8)

The reasoning that the high rate of auto accidents and the toxicity of arsenic, mercury and other toxic substances are "acceptable risks" to our society is justification for acceptance of a whole new source of possible contamination of the environment by radioactive substances is a specious argument. (195)

It is proper to compare, for example, the possible risks from coal ash disposal with possible risks from high-level and low-level nuclear waste disposal. However, it is improper to compare either of these risks with the risk of being hit by lightning or being injured in a highway accident because the risks are associated with wholly unrelated activities. (217)

Response

The discussion of risk and risk perspectives in the draft has been substantially modified and is presented in a separate section in the final Statement (3.4). The comparisons

RISK PERSPECTIVES

made in this section are for the purpose of giving a perspective as to the relative impacts of radiological wastes versus the hazards of other materials in the environment. Such comparisons are not intended to justify the potential impacts from radioactive waste management but are intended to assist the reader in understanding the concept of risk as it relates to radioactive waste management.

The points regarding chemical toxicity are well taken. The toxicity of many substances depends on chemical form whereas radioactivity is relatively independent of chemical form. However, some dependence still exists because the chemical state of the radionuclide determines its movement in the body.

The number of automobile deaths was included merely for perspective. A nation that apparently condones the loss of 55,000 lives annually on the highways may not, as one is sometimes led to believe, be interested in preserving human life.

Contamination of land is important as suggested. In the Statement the degree of importance is reflected in the number of health effects that such contamination would produce.

Draft p. 3.1.28

Issue

The measurement of the equivalence of the hazards of waste repositories and natural ore is a complex subject. Many of the hazard indices are concerned only with the amount of radioactive material and its toxicity, without consideration of routes by which the activity can reach man. (113-EPA)

Response

The comment is well taken. However, the routes to man for the ore and waste will be quite similar in this case.

Draft pp. 2.1.16, 2.1.18, 3.1.64

Issue

Several commenters noted that there does not appear to be treatment of the hazard index of the waste versus the toxicity of the natural ore (over time) in the context of the real risk presented by the spent fuel or high level waste in a repository. (38, 166, 196, 218-DOI)

Response

The high-level waste toxicity equals the toxicity of the ore that produced it at 1500 years. See Figure 3.4.1 of the final Statement.

RISK PERSPECTIVESDraft p. 3.1.64Issue

Paragraph 7 is very questionable. Hazard indices are not based on estimates of societal risks compared to other societal risks, in general. There is also the question of whether the hazard of the waste after several hundred years of decay, considering nuclides and pathways, is less than the hazard of the ores. (113-EPA)

Response

DOE disagrees. If one is to determine the significance of a hazard it must be compared to other hazards.

The text does not state that the hazard of the waste after several hundred years of decay is less than the hazard of the ores. The statement made was that the radioactivity of the wastes would not exceed that of the natural ore. However, because of the nature of a repository, i.e., 600 meters of overburden, compared to the overburden of most commercial mines, the hazard to the population from repository wastes stored several hundred years would not exceed the hazard from natural ores.

Draft p. 3.1.65, Hazard IndicesIssue

One commenter noted that this section contains several specious arguments culled from various sources and that more detail should be presented or the entire discussion be eliminated. (218-DOI).

Response

In preparing the final Statement an effort was made to provide more detailed information for the reader in the discussion of hazard indices (see Section 3.4).

Draft p. 3.1.65Issue

The following statement is made: "Eventually ^{239}Pu will disappear due to its 24,000 year half life; the lead will persist indefinitely." Although ^{239}Pu will decay, its daughter is ^{235}U which is the parent of a decay series terminating, after more than a dozen radioactive decays, in ^{207}Pb (lead-207). Thus the legacy of ^{239}Pu also persists indefinitely. (217)

RISK PERSPECTIVESResponse

Agreed, this statement has been changed in the final document.

Draft p. 3.1.65Issue

The total quantity of radioactivity in curies is irrelevant. The nature of the radio-nuclides and their pathways to man are significant. (113-EPA)

Response

The paragraph noted was poorly worded and has been deleted. The expression of quantity of radioactivity in curies without qualification is indeed irrelevant.

Draft p. 3.1.66Issue

Describe how the estimate of 1×10^{-6} for the "annual fatalities estimated due to isolated waste" is determined. Specifically, explain and justify the use of the "annual transfer probability for an atom of radium to enter the body from the geosphere." (208-NRC)

Response

Based on the quantity of ^{226}Ra in cadavers and the quantity of ^{226}Ra in the bio-sphere, the ratio of ^{226}Ra in man versus the biosphere was derived. This ratio, 4×10^{-13} , coupled with the quantity of radioactive material in the repository was used to calculate the dose to man. Based on the dose to the population, the health effects were calculated to be $1 \times 10^{-6}/\text{yr}$.

Draft p. 3.1.161Issue

The conclusion drawn from the comparison with the ore body would be improved if some analysis were provided. (113-EPA)

Response

An expanded treatment of deep geologic waste disposal and hazard from ore bodies appears in Section 3.4.1.

RISK PERSPECTIVESDraft Appendix HIssue

This appendix could well be omitted. Many of the hazard indices quoted are of no value as indices, and no information is available to enable one to select which, if any, of the indices are useful.

Although purporting to be a basis for determining the "hazard index," the material as presented in the appendix does not even approximate the potential hazard. The MPC is derived on the basis of dose to a "critical organ" rather than on the risk related to a given intake of isotope. The cumulative risk from intake of isotopes should be used as the basis for deriving a comparative "hazard index" since organ sensitivities are the controlling factor as noted in ICRP-26. (113-EPA)

Response

The intent was to indicate the variety of hazard indices that were available. Since there is no critical review of these indices then use is marginal. The revised appendix notes that--"Although each hazard index has merit for a particular set of conditions, the provision of simple measures of hazard can confuse rather than clarify. For this reason hazard indices are infrequently used in this Statement and dose and associated health effects are presented instead."

Issue

One commenter requested that the concentrations of plutonium in waste should be related to the natural radioactivity of the rock excavated for disposal. (6)

Response

For perspective on concentrations of plutonium the reader is referred to Figure 3.4.1. In the strictest use of concentrations (i.e. grams/cc of total rock) comparison of the plutonium content to say 0.2% uranium ore is not an improvement in clarity. This fails to recognize the need for pathways to man for the material to be significantly toxic.

Issue

A commenter stated that a mistake of the Statement was to rely solely on the technique of risk assessment as a criteria for safety decisions. (167)

Response

As noted in a previous response, impacts are presented from a consequence standpoint first. The document also identifies other criteria in Section 6.2 (in addition to radiological impacts) that the "decision-maker" would consider in evaluating the various disposal options.

RISK PERSPECTIVESIssue

Several commenters noted that the reader's perspective of relative risk would be further enhanced if health effects due to natural background were stated as a range rather than as a discrete value. (58, 124)

Response

By and large, reference to natural background as a comparison has been made on a basis of dose. Health effects for the regional population from natural background over a 70-yr period would (on the basis used in this Statement) range from 1,000 to 9,000.

Issue

Several commenters stated that a summary presentation should be provided for each option of all the elements of risk. (58, 124)

Response

From the standpoint of classical risk assessment, this issue calls for a detailed examination, and DOE does not feel that the data is available to make such a presentation with any confidence.

However, Chapter 7.0 of the final Statement does outline from a systems viewpoint the impacts of waste management (i.e., radiological and nonradiological impacts, land use, materials and energy requirements).

Issue

The Statement should specify incremental risk in initiating repository program in near-term versus deferring initiation until additional (or "complete" data) is obtained. (38, 134, 217)

Response

The Statement analyzed the environmental impacts of repository startup for the years 1990 through 2010 as part of the proposed action. The Statement also looked at repository startup in the year 2010 to 2030 and the associated impacts as part of the alternative action. The Statement did not perform a "risk calculation" per se comparing near-term repository startup versus a deferred startup.

Issue

Several commenters noted that the Statement should provide rationale for what is considered acceptable risk and/or define adequate safety. (6, 58, 97, 98, 124, 154)

RISK PERSPECTIVESResponse

EPA is currently formulating standards relative to radioactive waste management and disposal. Such standards are expected to include specifications concerning release rates to the accessible environment and will be used to determine the acceptability of a waste repository.

Issue

One commenter noted that a comparison of the potential for electrocution as a result of nuclear-generated energy and the hazards of commercial waste products be presented.
(178)

Response

There are about 55 linemen electrocuted in the U.S. each year bringing electricity to consumers regardless of the mode of electrical generation.

WASTE MANAGEMENT OPERATIONSReactor OperationsDraft pp. 1.7 and 2.1.2Issue

Several commenters noted that the assumed reactor performance characteristics assumed (40 year life and installed generating capacity) are difficult to justify in view of experience to date. (2, 40, 62, 181)

Response

With regard to reactor performance, DOE agrees that the operating experience with nuclear plants is not yet sufficient to predict their life-time performance characteristics with great accuracy. However, for purposes of the analysis of waste management, present assumptions are representative and considered adequate to predict potential effects.

Draft p. 2.1.2Issue

The impacts of an incorrect assumption regarding nuclear power plant performance degradation should be assessed in terms of how much less waste would be generated. (147)

Response

An analysis of five different nuclear power growth scenarios is provided in Chapter 7.0 of the final Statement. This chapter provides information on total energy generated for each growth scenario in terms of GWe-years and also the total quantities of waste generated. If plant performance is or is not degraded, the amount of waste generated will be proportional to the reduced or increased energy generation in each case.

Draft p. 2.1.10Issue

A burnup of 29,000 MWD/MTHM is low given that utilities are being given permission to use burnup to 58,000 MWD/MTHM. (55)

Response

If an average burnup higher than 29,000 MWD/MTHM is achieved, it will reduce the quantity of spent fuel. However, the total fission product activity would change very little while the actinide activity would increase as in the fuel recycle cases considered in this Statement. Thus, the waste disposal requirements arising from using fuel to higher burnup levels are within the envelope of the analyses presented in this Statement.

WASTE MANAGEMENT OPERATIONS

Draft p. 3:1.55

Issue

Because the effect of burnup on leach-rate has not been studied, the effect on leach rates of utilizing higher burnup fuel would be questionable. (55)

Response

DOE has established research programs to examine the impacts of higher burnup fuel.

Issue

The schedule for transferring spent fuel assemblies from power plants to disposition facilities should be presented. (129)

Response

A range of possible spent fuel transfer schedules is considered in the systems analysis in Chapter 7 of the final Statement.

Waste Characterization

Draft p. 1.1

Issue

One commenter requested clarification of the definition of "radioactive wastes" used in the draft Statement and noted that traditionally, high level waste (HLW) does not include transuranic (TRU) intermediate and low-level wastes. (208-NRC)

Response

DOE agrees that, traditionally, HLW does not include other TRU wastes. DOE cannot find (on cited page, at least) a statement that the TRU wastes are included in HLW. Section 4.2 of the final Statement describes the radioactive wastes with which this Statement is concerned.

Draft p. 1.5

Issue

One commenter objected to the confusion that can result when spent fuel is sometimes called spent fuel and other times called a radioactive waste. (181)

WASTE MANAGEMENT OPERATIONSResponse

DOE agrees that this can be a problem and has endeavored to be more precise in the final Statement. When discussing the once-through cycle, spent fuel is considered to be a waste, but in most cases we still refer to spent fuel as spent fuel or as spent-fuel waste. In the reprocessing cycle, spent fuel is not necessarily a waste because it contains the useful by-product uranium and plutonium but it also contains the waste products from the reactor operation.

Draft p. 1.7Issue

Sentence 3, paragraph 4 implies that there are high-level wastes from mixed oxide fuel fabrication. Is "spiked" fuel being referred to? This has not been indicated. (181)

Response

The high-level wastes referred to are generated in the fuel reprocessing plant and not the mixed-oxide fuel fabrication facility.

Draft p. 1.8Issue

One commenter requested a definition of transuranic (TRU) waste and wanted to know the concentrations of plutonium expected in TRU waste. (6) One commenter did not feel the use of 10 nanocuries per gram as definition of TRU waste is justified. (154)

Response

A definition of TRU waste is contained in the introductory paragraph to final Section 4.2. Variable concentrations of plutonium are expected in TRU wastes ranging from about 10^{-8} to 10^{-3} curies of transuranic alpha activity per gram of waste.

Draft p. 1.9Issue

The heat generation in HLW from one metric ton of spent LWR fuel is below 600 watts 30 years from the time of reprocessing, when heat generation was over 8 kw. The critical duration cited (500 years) is about an order of magnitude high when considering the impact of heat generation. (181)

WASTE MANAGEMENT OPERATIONSResponse

DOE agrees that the heat generation in high-level waste 30 years from the time of reprocessing is at or below about 600 watts per metric ton. However this does not mean that the heat generated following that time can be ignored or that the critical period is as short as 30 or 50 years. For a discussion of thermal criteria in a geologic repository and the temperature profiles in the geologic media as a function of time see Appendix K of the final Statement.

Draft p. 1.15Issue

Do sentences 1 and 2, paragraph 1, refer to intact spent-fuel assemblies, or to the HLW from the reprocessing of same? (181)

Response

Both. It is a summary statement concerning waste focus that may potentially be disposed of in geologic repositories.

Draft p. 1.17Issue

The term "fuel residues" is dangerously ambiguous. The commenter assumes that it was used to mean cladding hulls and other hardware. (181)

Response

The assumption is correct. DOE defined what it included as "fuel residues" in Section 3.1.4.2 of the draft Statement, but the comment arose from reading an earlier section. First use in the final Statement is in Section 4.2 where it is defined.

Draft p. A.3, 4, 20Issue

Several commenters questioned DOE's credibility for using "truncated lists" of radionuclides. The commenters claim that more than 90% of the known 400 radionuclides in the spent fuel are not accounted for. (42, 68)

Response

Appendix Tables A.2 and A.3 in the draft were intended to show only the most significant (in terms of radioactivity) nuclides in the waste components described. Table A.17 in

WASTE MANAGEMENT OPERATIONS

combination with Table A.16 and similar tables account for all of the radionuclides in the waste that have half-lives longer than a few weeks. While there are many more radionuclides that are initially present in spent fuel, many of those that are initially present have very short half-lives (measured in seconds and minutes) and are not important in terms of long-term waste disposal. The calculations carried out in support of this Statement considered 175 separate radionuclides. Not all of these were found to be significant in regard to disposal requirements. The tables in Appendix A identify quantities of 55 fission and activation products and 65 actinides and daughter actinides which for practical purposes are believed to account for all of the radioactivity in the disposal repositories. The detailed radioactivity Tables for the final Statement can be found in Appendix Tables A.2.1a-A.3.9b.

Draft p. A.17

Issue

Although not implicitly stated, it appears that the inventory values in Table A.14 were based on a charge of 3.8×10^5 MTHM. However, the mass associated with the ^{232}Th (+2 daughters) given in the 1,000,000 year column is 5.8×10^6 MT. There is an obvious error in the program used to generate this table. This single, obvious error brings into question all output generated by the computer program which was used to generate Table A.14. (208-NRC)

Response

The value for ^{232}Th is 1.91×10^1 Ci for $t = 10^6$ years. This error was not due to computer error but was a typographical error.

Issue

It would be helpful to summarize the important general characteristics of nuclear wastes. (6)

Response

The characteristics of nuclear wastes of concern to this Statement are now summarized in Table 4.2.1.

Issue

Popular misconceptions combine the short-term heat and high-level radiation properties of short half-life fission products with the long half-life of TRU wastes. Correct the misunderstanding by providing in the Summary a graphic representation of the decay of these properties with time. (13)

WASTE MANAGEMENT OPERATIONSResponse

This appears to be a good suggestion and plots are provided in the final Summary showing total radioactivity and total heat generation rate as a function of time for an example metric ton of spent fuel and a metric ton equivalent of high-level waste.

Issue

Detailed information on the nuclear growth scenario assumed should be provided, including; numbers and types of reactors that come on line each year; and the annual waste streams from the plants, including spent fuel and low-level waste (volume and activity). (208-NRC)

Response

Information pertaining to waste volumes and waste logistics is presented in Chapter 7.0 and Appendix A of the final Statement.

Issue

A concise summary of the chemical and biological properties of the important radionuclides in high-level waste should be provided, including solubilities and ion exchange characteristics with soil minerals. (6)

Response

While it is possible to concisely summarize radioactive half-lives and the types and energy of the radiation involved, a similar summary of solubility or ion exchange behavior is not practicable. These properties vary with temperature, water composition, and other factors, and are not amenable to being summarized concisely. Where these properties are important for specific analysis made in the Statement, the values used are stated concisely along with pertinent references.

Waste Treatment and PackagingDraft p. 1.9Issue

One commenter questioned the statement "in either event, the HLW contains fission products, uranium, plutonium, and the balance of the TRUs." In both the recycle options most of the uranium is removed, and in the U-Pu recycle most of the plutonium is removed as well. Furthermore, if one assumes U & Pu recycling, sooner or later one reaches the point where fuel elements no longer have sufficient fuel value to be worth recycling. This case should be considered.

WASTE MANAGEMENT OPERATIONS

The volatile materials and TRU elements separated in fuel reprocessing and captured in accordance with the uranium fuel cycle standards (40 CFR 190) are omitted in this discussion. They should be included. (113-EPA)

Response

The sentence in question does indeed appear to be redundant and need not have been included in the draft Statement. It does not appear in the final Statement.

DOE agrees that there is likely to be a point where the uranium and plutonium no longer have sufficient value to justify recycle. How much segregation of uranium and plutonium recycle batches may be practiced is an economic and technical decision that will be determined by the industry when and if recycle becomes a reality. Recycled uranium and plutonium are not necessarily contained in the same fuel elements. Segregation of recycled uranium will be difficult if it is sent back to the diffusion cascade for reenrichment. DOE has assumed in the system simulation (Chapter 7.0 of the final Statement) that uranium or plutonium enriched fuel elements that represent third recycle discharges will be reprocessed but neither the uranium nor the plutonium is further recycled. The uranium is assumed to be sent to storage with diffusion cascade tails for other potential uses. The plutonium is assumed to be dispersed in and immobilized with the high-level waste.

The handling of the volatile materials and TRU elements separated in fuel processing was included in the draft Statement in Section 3.1.4 and, in more detail, in Appendix L. In the final Statement, the treatment and storage of these materials are covered in Sections 4.3 and 4.4.

Draft p. 1.9

Issue

One commenter noted that regarding Table 1.1 of the draft Statement:

- Some of the numbers cannot be derived from other tables presented.
- Column heading or footnote should indicate that hulls and hardware are included in the TRU intermediate-level waste, if that is the case.
- Column heading should indicate that the low-level waste is TRU contaminated.
(208-NRC)

Response

Table 1.1 in the Summary of the draft Statement represented an attempt to present complex data-sets in a summary fashion, however, the table did contain errors and may have been overly condensed. It is not used in the final Statement.

WASTE MANAGEMENT OPERATIONS

To avoid confusion regarding the use of the terms intermediate-level TRU waste and low-level TRU wastes, DOE has adopted the convention in the final Statement of referring to these wastes as remotely-handled TRU waste and contact-handled TRU wastes.

Draft p. 1.9Issue

One commenter requested an explanation of the method used to obtain the values in Table 1.1 (208-NRC)

Response

This table does not appear in the final Statement. The basis for Table 1.1 can be found in Table 2.1.8 and Appendix A, Tables A.19, A.26, and A.35 of the draft Statement. Assumptions relative to waste treatment and treated waste volumes are discussed in Section 4.3 of the final Statement.

Draft p. 1.11Issue

Some sort of conversion factor should be provided so that realistic comparisons can be made between storage requirements for spent fuel and waste containers for reprocessed wastes. (147)

Response

Information on waste container requirements for treated wastes is provided in the final Statement in Section 4.3.6.

Draft p. 1.17Issue

The process of incinerating combustible waste has not been shown to be a safe method. (62)

Response

The final Statement includes references to the operation of combustible radioactive waste incinerators (see Section 4.3).

WASTE MANAGEMENT OPERATIONSDraft p. 1.18Issue

The fact that the wastes from MOX-FFPs are TRU wastes is more vital than their being "non-high-level." (181)

Response

DOE agrees with the statement, and has endeavored in the final Statement to use the term "TRU wastes" in preference to "non-high-level wastes."

Draft pp. 2.1.6, 8, 22Issue

Several commenters noted that the startup dates for reprocessing facilities are not realistic. (6, 35)

Response

The range of dates used in the analysis has been adjusted to the years 1990 to 2010. This range of dates was selected to bound the waste management impacts that one would expect from such facilities.

Draft p. 2.1.11Issue

One commenter asked if fuel reprocessing is to be done in DOE spent fuel reprocessing plants. (62)

Response

For this Statement, reprocessing of commercial fuel is assumed to be carried out at facilities under custody of the Federal government. These plants would have to be licensed for operation by the Nuclear Regulatory Agency.

Draft p. 2.1.11 and 2.1.13Issue

Cooling towers are listed as major environmental release points. However, sufficient engineering safeguards will exist to prevent releases through a closed cooling water system. (147)

WASTE MANAGEMENT OPERATIONSResponse

Cooling towers were mentioned as major environmental release points because DOE considers the environmental impact of nonradiological as well as radiological releases, and nonradiological releases do occur at cooling towers. DOE agrees that adequate technology exists to prevent radionuclide releases via cooling towers.

Draft p. 2.1.22Issue

One commenter noted that DOE should justify the assumption that spent fuel will be packaged prior to storage. This procedure should be based on a cost-effectiveness analysis. (208-NRC)

Response

The example case in the final Statement (see Section 3.2, 4.3 and 4.4) is now based on packaging spent fuel after storage and shows packaging prior to storage as an option including a generic cost for shipping packaged spent fuel. Selection of a procedure based on cost optimization is not yet feasible.

Draft p. 2.1.24Issue

No reason is given for packaging spent fuel assemblies while the option of reprocessing is still kept open. If reprocessing is the chosen option, sealed-steel canisters would be an economic and procedural burden at the head-end of the fuel reprocessing procedure, besides creating yet another waste stream. (181)

Response

DOE agrees and has revised the method assumed for storing spent fuel assemblies while the option of reprocessing remains open. Water basin storage of spent fuel is assumed for this interim storage. Storage of packaged spent fuel is considered a possibility only if a very long time of interim storage can be foreseen when it might be considered desirable to provide an extra measure of safety by storing the spent fuel in a packaged form.

Draft p. 2.1.26Issue

If the first reprocessing plant starts in the year 2010 some of the spent fuel may be as old as 50 years. What is the known experience of reprocessing 50-year old spent fuel?

WASTE MANAGEMENT OPERATIONS

Shouldn't that be considered in your GEIS? You tell us that the oldest fuel is reprocessed first. I assume you are proposing that the oldest fuel go first since you share some of my apprehensions. (55)

Response

If reprocessing is delayed for many years, it is conceivable that some of the spent fuel may be as old as 50 years before reprocessing. No problems are foreseen in reprocessing this age fuel. In fact, it should be considerably easier to reprocess 50-year old fuel than much younger fuel because the radioactivity will be substantially reduced. All of the short half-life nuclides will have decayed to insignificant levels. Most of the tritium and krypton will be gone and the amount of ^{90}Sr and ^{137}Cs will be reduced by nearly 70%. See Section 4.3 of the final Statement.

Draft p. 3.1.54, Item 3

Issue

Unless actinides from chemically separated high-level waste are recycled, they must be disposed of as waste and would still require consideration in this Environmental Impact Statement. Even if one assumes recycling of uranium and plutonium, one eventually reaches the point where recycling is not economically feasible and the transuranics must then be disposed of as waste. (113-EPA)

Response

This comment concerns a statement that was apparently interpreted to mean that not all of the actinides in the spent fuel are sent to the repository. This appears to be a case of misinterpreting the intent of the text in the draft Statement. In the reprocessing fuel cycle, all of the actinides except for plutonium are sent to the geologic repository with the high-level waste. DOE agrees that one may eventually reach a point where recycling is not economically desirable. In the system simulation in Chapter 7.0 of the final Statement, DOE assumes that plutonium discharged from the third recycle is no longer attractive as a recycle fuel and it is discarded to the high-level waste.

Draft p. 3.1.84, Table 3.1.4

Issue

Wastes from spent fuel cleaning operations should be included in intermediate or low-level wastes listed in this table. (58)

WASTE MANAGEMENT OPERATIONSResponse

While wastes from spent fuel cleaning operations would indeed be expected to be radioactive, they are not expected to contain sufficient transuranic radioactivity to require consideration as transuranic waste. Since the Statement is not concerned with treatment of nontransuranic wastes, these wastes were not included in the table.

Draft pp. 3.1.84 and 3.1.186

Issue

Table 3.1.4 compares volumes of wastes from four possible fuel cycles. Text should note that volumes differ by no more than a factor of three. The footnote to the table and the assumption on p. 3.1.186 which indicates that 38 MTHM of reference spent fuel and 7.6 MTHM MOX fuel produced per GWe-yr is 50% too high. (154)

Response

DOE agrees that the volume ratio of 3:1 is a valid observation. See Section 3.4.6.

38 MTHM per GWe-yr is correct. The average fuel exposure for the reference fuel is approximately 29,000 MWD/MT. Thus:

$$\frac{365 \text{ D/yr} \times 1,000 \text{ MW/GW}}{29,000 \text{ MWD/MT} \times 0.33 \text{ MWe/MW}_{\text{Th}}} = 38 \text{ MTHM/GWe-yr}$$

The author of the comment probably had in mind the amount of fuel discharged from a 1,000 MWe plant which, because it operates at say 70% of capacity over the period of a year, produces 0.7 GWe-yr of electricity and discharges about $0.7 \times 38 = 27$ MTHM. Each metric ton of spent fuel can provide plutonium for approximately 0.2 metric tons of MOX fuel thus 7.6 MTHM MOX fuel per GWe-yr.

Draft p. 3.1.87

Issue

One commenter questioned why plutonium would be partially purified before blending with the high-level liquid waste (in one of the uranium-only recycle cases considered in the draft Statement). (181) Another commenter suggested that it may be done for reasons of criticality. (154)

Response

There is no criticality reason requiring the partial purification of plutonium prior to putting it back into the high level waste; DOE did not intend the wording would be taken to suggest the opposite. What was meant is that liquid high-level waste containing all of

WASTE MANAGEMENT OPERATIONS

the plutonium could not be stored in large tanks because of criticality reasons, so the plutonium was recovered separately, and then blended back just prior to solidification. Partial purification occurred during the recovery, but it was not required for criticality reasons.

Draft p. 3.1.87

Issue

One commenter expressed concern over the concept of storing aqueous plutonium nitrate for "several years" prior to blending it into high-level liquid waste for solidification in one of the uranium-only recycle options. (181)

Response

In the discussion of the uranium-only recycle option in the draft Statement, the approach selected as a reference involved essentially immediate solidification of the high-level waste and plutonium nitrate mixture so that very little storage of plutonium nitrate solution would be required.

Draft p. 3.1.87

Issue

The technique of using concrete to consolidate hulls in 55-gallon drums should be mentioned. (181)

Response

This technique is included in the final Statement (see Section 4.3.3.1).

Draft p. 3.1.87

Issue

Calcination (conversion of metals in soluble salt solutions to dry, metallic oxides) can hardly be called an alternative solidification process. It's the first step or co-step in all consolidation processes. (181)

Response

Section 4.3.2.3 of the final Statement points out that various consolidation techniques can be utilized following calcination.

WASTE MANAGEMENT OPERATIONSDraft p. 3.1.90Issue

To say that "all" the fission product tritium is released in reprocessing is not true. A large proportion of the tritium is bound to the Zircaloy cladding as the hydride. Only if the fuel is first roasted, as in voloxidation, will the hydride decompose, oxidize to water, and thus be removed from the cladding. (181)

Response

It was noted in the draft Statement, "the reference FRP operation results in the release to the atmosphere of essentially all the tritium present in the spent fuel." DOE meant the fuel itself (UO_2), but can see now that the wording could be construed as including the cladding. The wording in the final Statement (Section 4.3.4.2), "...the tritium present in the UO_2 portion of the spent fuel is released...", should be clearer.

Based on discussions with personnel at Oak Ridge National Laboratory, the tritium bound to the cladding will be released during voloxidation only if a temperature higher than that currently planned for use with LWR fuels is employed.

Draft p. 3.1.90Issue

Several commenters noted that the storage and separation of ^{85}Kr may create a greater hazard than allowing it to disperse and the rationale for ^{85}Kr storage should be given. (6, 35, 154) One commenter recommended decay interval storage of ^{85}Kr . (28)

Response

Direct release of ^{85}Kr versus its separation and storage did receive additional discussion in the final Statement (see Section 4.4.4 and 4.7).

The rationale for the storage of ^{85}Kr for 50 years before it is released is that the radioactive decay reduces the amount present after 50 years to only about 4% of that present initially; a factor of 25 reduction in the amount released is thus achieved. The 50-year storage period is not necessarily optimum. However, it is well within our technical capability, and reduces total krypton released to within EPA standards which limit ^{85}Kr releases to 50,000 Ci/GWe-yr (10 CFR 190.10).

Draft p. 3.1.191Issue

The quantity of ^{85}Kr released to the air should be related to gigawatts of electricity produced for comparison with uranium fuel cycle standards. (113-EPA)

WASTE MANAGEMENT OPERATIONSResponse

This has been done. See preceding response.

Draft p. 3.1.191Issue

Table 3.1.68 appears to be in error with regard to ^{14}C . ^{14}C is supposed to be removed in reference FRP. The ^{14}C value shown is the full amount in the fuel. (154)

Response

^{14}C in Table 3.1.68 was given incorrectly. The value should have been given as and was corrected to $2 \times 10^1 \text{ Ci-}^{14}\text{C}$.

Draft p. A.58Issue

Table A.52 shows 5,760 metric tons of plutonium in spent fuel in the U + Pu recycle mode. Our calculations indicate that this quantity of plutonium indicates an extremely high mix of spent fuel from plutonium recycle as compared with UO_2 -enriched uranium only fuel. As averaged over the entire time span to the year 2040, the MOX to UO_2 fuel ratio we calculate is 60/40. Please provide your basis for this estimate. (208-NRC)

Response

The data in Table A.52 in the draft Statement were developed using the computer code ENFORM (see Section 7.2 of the final Statement). This program calculates fuel cycle logistics and recycles recovered plutonium as MOX fuel as it becomes available. Table A.52 shows 5,760 metric tons of plutonium in spent fuel but it also shows 4,400 metric tons of plutonium actually recycled. The ENFORM program results for this case show 97,000 MT of MOX fuel and 379,000 MT of total fuel. This is a 26/74 ratio of MOX to UO_2 enriched fuel. The recycled plutonium is approximately 60% fissile plutonium. Thus, the 60/40 ratio calculated by the author of this comment contains a substantial error.

Draft Appendix NIssue

One commenter felt that some discussion should be included concerning the useful life of spent fuel casks and noted that the analysis appears to assume the casks used in the early time frame will be available for use 30-40 years later. (208-NRC)

WASTE MANAGEMENT OPERATIONSResponse

The useful life of spent fuel casks was estimated in DOE/ET-0028 (Section 6.2) to be 20-30 years. A 10 year capital recovery period was used in the cost calculations but an indefinite life was used when estimating the resource requirements. In the final Statement a 20-year life is used to estimate resource requirements.

Draft p. N.2Issue

One commenter noted that the cask availability data in the draft Statement is out of date. The commenter also requested that some discussion be provided relative to the industry's ability to meet demand for spent fuel casks at the rate they will be required. (208-NRC)

Response

The cask availability data have been updated for the final Statement in Section 4.5, Table 4.5.1. The second issue is addressed in DOE's EIS on Spent Fuel Policy (DOE 1980b) in Volume 2, Section III.

Draft p. N.7Issue

One commenter questioned why the cask maximum thermal design load is set at 50 kW (208-NRC).

Response

The conceptual high-level waste shipping cask described in this report is based on work done by Battelle-Northwest (Rhoads 1978). The 50 kilowatt heat rejection limit is imposed by the design considered in this report. Nine canisters of 5-year-old solidified high level waste, 30 cm in diameter and 3 meters long, could be carried by this design.

DOE/ET-0028, p. 4.1.30Issue

In DOE/ET-0028, supporting documentation for the Statement, fluidized bed calcination is identified as the most well developed calcination process, yet the reference vitrification process is spray calcination/in-can melting. Why is one calcination process referenced to make glass waste form and another to make calcine waste form? (208-NRC)

WASTE MANAGEMENT OPERATIONSResponse

The selection was based on degree of development and amount of operating experience for the given service. Fluidized bed calcination has been used to calcine radioactive waste at the Idaho National Engineering Laboratory since 1963. This operating experience made it the obvious choice for the reference process to make a calcine waste form. However, there has been only very limited experience in coupling a fluidized bed calciner to a glass melter. Instead, spray calcination has been quite well developed for use with waste glass melters in the U.S. and in Germany, therefore it was selected for the reference vitrification process.

Waste StorageDraft p. 1.1Issue

One commenter requested that the Statement should include interim storage facilities in the general description of the fuel cycle since it is apparent from the discussions in the Statement that these facilities will be built. (208-NRC)

Response

DOE agrees with the concept stated here. Interim storage facilities are described in Section 4.4 of this final Statement. The number and type of interim storage facilities that "will be built" depends on the fuel cycle that is implemented, processing mode chosen for the fuel cycle, and also on the timing of repository availability relative to treatment of the wastes. For a discussion of storage requirements with a range of fuel cycle assumptions see Section 7.3 of the final Statement.

Draft p. 2.1.3Issue

The statement, "Storage can occur either at the reactor site or at an offsite independent spent fuel storage facility (ISFSF) sometimes referred to as away from reactor (AFR) storage," is misleading. The fact that an ISFSF can be an at-reactor or an AFR spent fuel storage site was not made clear to the public in the draft Statement. (55)

Response

DOE agrees that the quoted sentence does not make clear to the public that spent fuel can be stored in an ISFSF, either at the reactor site or away from the reactor. DOE felt that the important point was that spent fuel can be stored either at the reactor or at an

WASTE MANAGEMENT OPERATIONS

AFR. DOE has clarified in the final Statement, Section 4.4.1.1, that storage at a reactor site may be in either the reactor storage basins or in an ISFSF located on the same site as the reactor.

Issue

Several commenters questioned the water basin storage period for spent fuel.

Draft p. 2.1.4--There are some apparent inconsistencies in sections on spent fuel cooling times and age at the time of disposal. There is not reasonable chance for disposal at 6.5 years of age and a more likely age is 20 to 25 years. For these reasons the sizes of repositories have been significantly overstated. A minimum cooling time prior to disposal of 20 years is proposed for the final Statement and it is suggested that a reasonable option is further cooling for another 20 to 40 years. (154)

Draft p. 3.1.83--What is the reason for the assumed 6.5-yr storage period, and without some kind of explanation, 6.5 years seems arbitrary. (181)

Draft p. 3.1.173--Cooling only 6 to 7 years before disposal will not occur in the foreseeable future. (154)

Response

DOE agrees that the age of the spent fuel or high-level waste sent initially to the first repository will be much greater than 6-1/2 years, because of the large backlog of spent fuel that will accumulate by the time the first repository becomes available. The age of the waste at the time of disposal is discussed in Section 7.3.4 in the final Statement. The minimum spent fuel age for the earliest repository date is approximately 15 years and ages up to 55 years are shown to be conceivable. However, it is assumed that the storage backlog will eventually be worked off and the minimum design age of 5 years for spent fuel and 6.5 years for solidified high-level waste will eventually be reached. DOE also agrees that it is possible that a decision may be reached to continue to provide interim storage so that the minimum age for delivery to a repository might be maintained at 20 years or greater. However, for the final Statement, it was considered prudent to assume the possibility of younger waste going to the repository so that the impacts would not be understated.

Draft pp. 2.1.22, 2.1.25, 3.1.184, 3.1.181, 3.1.184, 3.1.186

Issue

One commenter noted that it appears that all terms listed below refer to the same facility. Terms should be used consistently throughout to avoid confusion and to facilitate comparisons. (208-NRC)

WASTE MANAGEMENT OPERATIONS

Packaged spent fuel storage facility
 Storage
 Offsite storage facilities
 Extended storage facilities
 Storage facility
 (ESFSF) Extended Spent Fuel Storage Facilities
 Dry caisson storage facility
 SURF

Response

DOE agrees it is desirable to use terms consistently throughout to avoid confusion and to facilitate comparisons. DOE also agrees that some valid confusion over terminology could have resulted in the draft Statement; efforts have been made to correct this in the final Statement. The acronym AFR has been used in the final Statement to denote a storage facility separate from the reactor site.

Draft p. 2.2Issue

Four impact statements on TRU waste are mentioned as being in preparation by DOE (SRP, INEL, RL, AND LASL). Data from DOE received by NRC in conjunction with the DOE licensing study showed TRU waste to exist at ORNL. Will there be an environmental statement for ORNL? (208-NRC)

Response

Yes, Oak Ridge National Laboratory is also preparing an EIS for transuranic wastes.

Draft p. 3.1.83Issue

Spent fuel storage at the reactor pool certainly will be done, not "can be done," for a minimum of six months. (181)

Response

The referenced statement in the draft Statement was, "...interim storage of the unpackaged fuel can be done at either the power plant basin or at an independent spent fuel storage facility..." DOE did not intend to imply that the duration of storage at the reactor would be less than six months, which appears to be what concerned the commenter. DOE agrees that spent fuel will most likely be stored for at least six months in the reactor pools.

WASTE MANAGEMENT OPERATIONSDraft p. 3.1.85Issue

The draft Statement is inconsistent in saying one place that spent fuel assemblies can be stored dry in the unpackaged condition, but saying in a second place that dry storage of unpackaged fuel is not feasible. (181)

Response

The draft Statement is not inconsistent in this instance; rather, it contains a distinction that apparently was missed by the reviewer. In the one place, DOE said that spent fuel assemblies can be stored in the unpackaged condition in forced-draft air-cooled vaults. In the second place we said that natural-draft air-cooled vault storage was not feasible for unpackaged fuel. The distinction is drawn because of the belief of DOE that the effluent air from an unpackaged fuel storage facility should be filtered before it is released to the atmosphere. The pressure drop attending the use of appropriate filters does not allow natural-draft cooling to be employed. In the interest of brevity, this distinction was not detailed in the draft Statement.

Draft p. 3.1.85Issue

Interim storage of packaged spent fuel will increase the danger of leaking. Note the "leak after leak at Hanford, West Valley, Savannah." (30)

Response

DOE feels confident that packaged spent fuel storage facilities can be operated with negligible release of radionuclides to the environment. It should be noted that these sites have experienced some leaks from liquid waste storage tanks, not from packaged spent fuel storage facilities (which have not been employed at these sites).

Draft p. 3.1.92Issue

Use of "refrigerated air" to cool Kr-storage cylinders implies installation of a refrigeration system which is, based on the low heat load from the cylinders, both unnecessary and impractical. (181)

Response

The Kr-storage cylinders each have an initial heat generation rate of 150 W. In the storage concept considered here, 104 cylinders are contained in a storage cell and

WASTE MANAGEMENT OPERATIONS

recirculating, refrigerated air is provided for each cell to remove the heat and to provide improved leak detection and containment in case of leaks. Other storage concepts could doubtless be developed in which the use of refrigerated air would be unnecessary.

Draft p. 3.1.92

Issue

If only 10% of the krypton is released in the dissolver, where does the rest go? Would it not be released during shearing and dissolution? (181)

Response

DOE agrees that all of the krypton would be released to a gas phase during shearing and dissolution. The statement that, "Ten percent of the inventory of ^{85}Kr is released in the dissolver off-gases." was meant to convey the information that we assumed that 10% of the ^{85}Kr was not removed by the dissolver off-gas treatment process and was released to the atmosphere.

Draft p. 3.1.178

Issue

Where do the releases in draft Table 3.1.58 come from? Why is there a factor of 50 difference in routine releases to air between ^{137}Cs and ^{90}Sr ? (154)

Response

The releases in draft Table 3.1.58 are based on releases during transportation, receipt, storage and packaging of spent fuel based on data contained in the support documents DOE/ET-0028 and -0029. The difference in release fractions for ^{137}Cs and ^{90}Sr are due to the release fractions contained in DOE/ET-0028. These release fractions are based on the higher leachability of cesium compared to strontium. This causes the larger release fraction for cesium which results in larger source terms and larger doses.

Draft p. 3.1.184

Issue

One commenter suggested that the document provide support for the statement that no releases of radioactivity would occur during planned operation of the extended spent fuel storage facility (dry caisson options). (208-NRC)

WASTE MANAGEMENT OPERATIONSResponse

This matter is now covered by the statement that double containment of the radionuclides is depended on to maintain releases at negligible levels (see Section 4.4.1.4 of the final Statement).

Draft p. 3.1.186

Issue

One commenter disagreed with some of the reprocessing case assumptions stating that they seriously overstate source terms and thus environmental effects. The following basis was suggested.

1. 30 tonnes fuel per GWe-yr
2. Shipment to AFR at 6.5 years
3. Reprocessing with 15 years cooling
4. Disposal to repository with 20 year cooling. (154)

Response

1. In the final Statement, DOE assumes 7 years capacity at the reactor basins after which spent fuel is shipped to AFR's.
2. For the earliest repository availability (1990) the age of fuel at reprocessing is approximately 13 years but as the storage backlog is worked off, this declines to 1.5 years.
3. An additional 5 years cooling for the high-level waste after reprocessing is assumed. Thus, the age of the initial waste for the earliest repository is approximately 18 years. This declines to 6.5 years after the storage backlog is worked off.

Draft p. A.49

Issues

Explain the erratic discharge schedule for spent fuel in the years between 1983 and 1988 that is indicated by the inventory accumulation. (208-NRC)

Response

Table A.43 is a typewritten table and the problem is the result of a typographical error. The table shows 13,600 MTHM of spent fuel in reactor storage basins in both 1985 and 1986. The 1985 value should be 12,900. Use of this value will eliminate the erratic discharge schedule observed by the commenter.

WASTE MANAGEMENT OPERATIONSDraft p. N.1Issue

One commenter pointed out a perceived inconsistency to the effect that in one location (Table N.1) the draft Statement does not show movement of spent fuel from reactor directly to reprocessing plant (which would occur for recycle options) but in another location (p. 2.1.5) the draft Statement says that storage requirements can be met by power plant storage basins for the recycle options (thus allowing direct shipment to reprocessing). (208-NRC)

Response

This perceived inconsistency apparently arose because a qualifying statement in the first location was overlooked. This statement was "...wastes that are assumed to require Federal custody for storage or isolation..." If spent fuel is going to a reprocessing plant, DOE did not assume it to require Federal custody. This material has been completely revised in the final Statement. Final Section 3.2.1.2 discusses the movements of spent fuel in the reprocessing cycle and these movements are illustrated in Figure 3.2.2. Additional details on spent fuel shipments such as that contained in Appendix N of the draft Statement are presented in Section 4.5.1 in the final Statement.

Issue

One commenter requested that the basis be given for assuming that two Independent Retrievable Waste Storage Facilities would be needed to serve the needs of the reprocessing industry if repositories are not available until the year 2000. Particular interest was expressed in the economics of facility costs versus transportation costs. (208-NRC)

Response

The requirement was based solely on the quantity of waste requiring storage and the assumed capacity of a retrievable waste storage facility for reprocessing and MOX plant wastes. The capacity of a single facility was arbitrarily set to meet the storage requirements at the year 1995. This was equivalent to the wastes resulting from processing 45,000 metric tons of spent fuel. In the scenario employed in the draft Statement the requirement increased to 77,000 metric tons equivalent by the year 2000. Thus two retrievable waste storage facilities were required. In the final Statement, five different nuclear growth scenarios are considered. Storage requirements for these cases with two different reprocessing dates and three different repository dates are described in Chapter 7.0 of the final Statement.

WASTE MANAGEMENT OPERATIONSWaste Transportation

Draft pp. 1.11

Issue

The transportation operation should receive further attention because at all points along the routes the general population and the operating personnel would be exposed to above ambient levels. (144)

Response

As Section 4.5 points out, the transportation system must be designed to be consistent with the regulations outlined in Title 49 and Title 10 of the Code of Federal Regulations. The prescribed exposure limits for both the general population and operating personnel cited in the above documents are lower than ambient levels.

Draft p. 3.1.85

Issue

There is confusion over the meaning of three sentences regarding the shipping of spent fuel (packaged or unpackaged) between wet and dry storage facilities and to the repository. (181)

Response

DOE feels that much of the confusion on the point of the reviewer comes from our efforts to be brief and yet also at least mention the various options that can be implemented. DOE has tried to minimize such confusion in the final Statement, where spent fuel packaging, storage, and transportation are addressed in Sections 4.3.1, 4.4.1, and 4.5.1, respectively.

Draft p. 3.1.93

Issue

Nonrequirement of neutron shielding for shipping cask hulls implies that hulls do not generate neutrons, which is not the case. (181)

Response

This implication was not intended.

WASTE MANAGEMENT OPERATIONS

Draft p. 3.1.175

Issue

Note the assumption that 90% of fuel from reactors to ISFSF and 100% from ISFSF to Repository goes by rail. This is not likely. This assumption leads to 89,000 truck shipments totalling 280 million km (1.7E+08 miles). This is equivalent to 2.5 highway fatalities, which would become 25 highway fatalities if all trucks were used as is more likely. (154)

Response

Because of the great advantage to receiving and loadout efficiencies when handling large rail casks compared to handling 5 to 10 times as many trucks casks and because of reduced radiation exposure to the public during the transport, DOE assumed that all shipments between major waste management facilities such as independent spent fuel storage basins and geologic repositories would be 100% by rail. Because many reactors do not have rail facilities, we assumed that 10% of the fuel shipments from reactors would be by truck and one-half of the remaining 90% that is shipped by rail would be shipped in intermodal casks allowing shipment by truck to the nearest rail head.

Draft p. N.2

Issue

The document states that about 50% of the operating reactors do not have rail spurs at the site. The reference system given on p. N.3, line 5, shows 90% by rail and 10% by truck. Is this 50% by rail, 40% by intermodal rail and truck, and 10% by truck? Note: on p. N.5, a 45%/45%/10% breakdown is given. (208-NRC)

Response

The fact that 50% of operating reactors do not have rail spurs is factored into the analysis. The 45%/45%/10% breakdown given on draft page N.5 refers to the type of casks used; 45% NLT 19/24 (rail cask), 45% IF300 (intermodal truck/rail cask) and 10% SNF4 (truck cask). This allows as much as 55% of the shipments to start out by truck but 45 of 55 are transferred to rail cars at the nearest rail siding. Thus the 90% rail 10% truck division.

Draft p. N.3

Issue

The impacts presented in Table N.3 and N.4 of the GEIS are based on a shipping scenario where 100% of all shipments are transported either by rail or by truck. Whether these impacts are presented only for comparison purposes or whether the scenarios upon

WASTE MANAGEMENT OPERATIONS

which these impacts are based are alternatives to be considered in addition to the reference case is not clear. If the latter is true, then the impact of building rail spurs to the 50% of reactors that do not have these spurs should be given in the GEIS. For the reference case, the impact of transporting the spent fuel by truck from these reactors to the nearest rail siding does not appear to have been included in the analysis. (208-NRC)

Response

The impact of 100% shipment by rail or truck are presented for comparison purposes only and are not alternatives to the reference case. For the reference case, the impact of transporting the spent fuel by truck to the nearest rail siding has not been included in the analysis.

Draft pp. N.3 and N.4Issue

Impacts presented on p. N.3 (Tables N.3 and N.4) and on p. N.4 are based on the assumption that all spent fuel is shipped by either rail or truck. Values given in DOE/ET-0029 are based on the reference case of 90% of the spent fuel being shipped by rail from reactors to ISFSFs and 10% by truck with 100% of the shipments from ISFSFs to the final repository being transported by rail. We recommend converting the results presented in Table N.3 and N.4 and on page N.4 to the reference case so that actual resource commitments can be known and comparison of the GEIS with the backup documentation can be facilitated. (208-NRC)

Response

Since this reference case is already presented in DOE/ET-0029, the corresponding portion of Appendix N is not required and was therefore deleted. See DOE/ET-0029.

Draft pp. N.3 and N.4Issue

It is not clear that the impacts shown in the GEIS have been correctly obtained from DOE/ET-0029. The following discussion develops ratios which can be applied to the results in DOE/ET-0029 to convert them into results that would be obtained if 100% of all shipments are transported by either rail or truck. Following this ratio development discussion is a table outlining some cases where impacts presented in the GEIS appear to have been improperly obtained from DOE/ET-0029.

Table 26.2.3 of DOE/ET-0028 shows 7,370 packaged PWR assemblies and 11,340 packaged BWR assemblies needed shipment from ISFSF to a final repository in the year 2000. In the reference case these assemblies would be shipped by rail in a modified NLI 10/24 cask which can

WASTE MANAGEMENT OPERATIONS

accommodate only 7 packaged PWR assemblies or 17 packaged BWR assemblies. (Normally, this cask can handle 10 PWR or 24 BWR assemblies.) For truck cask normally only 1 PWR and 2 BWR assemblies can be accommodated. Assuming a modified truck cask can be developed that can accommodate 1 packaged PWR assembly or 1 packaged BWR assembly, the number of truck shipments, for the year 2000, from an ISFSF to a final repository would be $7,370 + 11,340 = 18,710$ truck shipments. Table 4.1.1-3 of DOE/ET-0029 indicates a ratio of 120,000 to 1,700 for the total number of shipments through the year 2050 compared to the number for the year 2000. Applying this ratio to the 18,710 truck shipments results in a total of about 1.3×10^6 truck shipments through the year 2050 for movement of packaged spent fuel from and ISFSF to a final repository. To determine that total number of truck shipments, the number of truck from reactors to ISFSFs must be added to this value of 1.3×10^6 truck shipments.

Table 4.1.2-1 shows 8.9×10^4 truck shipments through the year 2050 for the reference system. Since this reference system is based on only 10% of the reactor shipments being transported by truck, a total of 8.9×10^5 truck shipments would occur if 100% of the shipments were transported by truck. Thus, the total number of truck shipments of all types through the year 2050 would be $1.3 \times 10^6 + 8.9 \times 10^5 = 2.2 \times 10^6$ truck shipments. Impacts presented in the GEIS for 100% of all shipments by truck should be $(2.2 \times 10^6) - (8.9 \times 10^4)$ or 25 times greater than the impacts given in DOE/ET-0029.

A ratio can also be developed for rail shipments. The reference system has 100% of shipments from ISFSFs to the final repository being transported by rail and no conversion to a 100% rail system is needed here. For shipments from reactors to ISFSFs, the reference system has 90% of all shipments transported by rail. Table 4.1.1-3 of DOE/ET-0029 shows 89,000 reactor shipments, through the year 2050, transported by rail. Since this is 90% of all shipments transported by rail, an all rail shipment scenario would have about 12,000 rail shipments. To this total must be added the 12,000 rail shipments from ISFSFs to final repositories, also shown in this table, for a total of 219,000 shipments for an all-rail scenario. For the reference system, the total number of rail shipments is $89,000 + 120,000 = 209,000$ shipments. Thus, the ratio of the number of rail shipments for an all-rail shipping scenario to the number of rail shipments in the reference scenario is $219,000 / 209,000 = 1.05$. Impacts presented in the GEIS for rail shipments should therefore be 1.05 times greater than impacts given in DOE/ET-0029.

A comparison of some of the GEIS results with those presented in DOE/ET-0029 indicates that the ratios developed in the above discussion are apparently the values used in converting impacts from one document to another. For example, on page 4.1.15 of DOE/ET-0029, Table 4.1.2-3 gives a value of 3.1×10^2 man-rem for the dose to the population living along the transport route, through the year 2050, for spent fuel truck shipments. Using the ratio derived in the above discussion, the impact presented in the GEIS, for an all-truck shipping scenario, should be 25 times greater giving a value of 7.8×10^3 man-rem

WASTE MANAGEMENT OPERATIONS

and indeed the result given in the GEIS is 8×10^3 man-rem. There are some values, however, that do not agree after this ratio is applied. Cases where there is a lack of agreement between the two documents are outlined in a table which follows the discussion of rail shipments.

For rail shipments, it is more difficult to determine if results for the reference system given in DOE/ET-0029 have been properly converted to an all-rail system which is used as the basis for impacts in the GEIS. The difficulty arises because the two systems are so similar and only differ by the 10% of shipments from reactor to ISFSFs that are transported by rail. It is difficult to determine if the ratio of 1.05 derived above has been used or whether a ratio of 1.11 has been used. The 1.11 ratio is obtained from the fact that the reference system has 90% of shipments from reactors to ISFSFs transported by rail, and this may have been improperly applied to the total system to include shipments from ISFSFs to final repositories which for both systems are 100% by rail. In addition, both ratios are close to 1.0 and some results presented in the GEIS have been rounded off, making it difficult to determine which ratio, if any, has been used. For example, the amount of diesel fuel needed through the year 2050 is given in Table 4.1-5 of DOE/ET-0029 as 1.7×10^6 m³. On page N.3 of the GEIS, it is stated that 2×10^6 m³ of diesel fuel is needed for an all-rail shipping scenario. It is therefore difficult to determine what, if any, ratio was applied to obtain this result. The following table outlines cases where the impacts presented in the GEIS are substantially different than properly converted values obtained from DOE/ET-0029. Values given in parentheses are the results that would be obtained if DOE/ET-0029 values are multiplied by the appropriate conversion factor developed in the above discussion, i.e., 25 for truck shipments, 1.11 for rail shipments. It should be noted that there is one impact where apparently the incorrect ratio of 1.11 was used instead of 1.05. This is the result for nonradioactive effluents released through the year 2050 for spent fuel rail shipments. Table 4.1.1-6 of DOE/ET-0029 shows, for example, that 4.8×10^3 MT of particulates will be released under these circumstances. Applying the incorrect ratio of 1.11 gives a result of 5.3×10^3 MT and this agrees with the result presented in Table N.4 of the GEIS. If the proper ratio of 1.05 had been used, the GEIS result would be 5.0×10^3 MT. Since the results are not substantially different and are within the uncertainty of these types of calculations, improper conversions of this type are not included in the following table. It is recommended, however, that for accuracy and consistency, the values given in the GEIS be properly converted. (208-NRC)

Response

All numbers were verified and errors in the Statement have been corrected.

WASTE MANAGEMENT OPERATIONSDraft p. N.5Issue

One commenter requested that some discussion be included describing the composition of a special train and the advantages and disadvantages resulting from its use. Is it a safer mode of transport? Does it have better safeguard features? (208-NRC)

Response

Special trains were assumed not to be used in the spent fuel transportation for this report. Increased safety by using special trains has not been demonstrated (Rhoads 1977). Disadvantages of special trains are their high cost. Details of cost aspects may be found in Section 6.2.1.6 of DOE/ET-0028. Some discussion is given in final Section 3.2 on the effects of interim NRC safeguard rules for the shipment of spent fuel.

Draft p. N.7Issue

One commenter felt that the discussion in the draft Statement which pointed out that heat dissipation is a prime consideration in the shipment of spent fuel may be misconstrued. The prime safety considerations are containment, shielding, and subcriticality. Heat dissipation is important to the performance of the other safety features. (208-NRC)

Response

Use of the word "prime" may indeed have been in error and "important" would have been a better choice.

DecommissioningDraft pp. 3.1.86 and 3.1.87Issue

The discussion on decommissioning should draw clearer distinctions between decommissioning requirements for reactor plants and the other fuel cycle facilities (since neutron activation of equipment does not occur in the other facilities). What isotopes are of interest at storage facilities? (181)

Response

The decommissioning section of the final Statement (Section 4.6) does not address reactor plants (since they are not a part of high-level or TRU management activities). Data

WASTE MANAGEMENT OPERATIONS

are now presented on the isotopes present in the decommissioning wastes from FRPs and MOX-FFPs and on the quantities released to the environment as a result of the decommissioning activities.

Draft Appendix DIssue

The 1,000 year storage and surveillance assumptions used in the calculations are in conflict with proposed Criteria for Radioactive Wastes (43 F.R. 53262 et seq., November 15, 1978) developed by EPA. The appendix should be revised using the proposed period of storage and surveillance of no more than 100 years. (113-EPA)

Response

Two decommissioning alternatives were analyzed for each of the three basic fuel cycle facilities considered in the Statement. The alternatives were selected to show the possible range of environmental effects from decommissioning activities. The alternatives considered are permitted under current regulations, although it is recognized that future regulatory activities by NRC, EPA, and state agencies could significantly impact the choice for decommissioning alternatives at a particular facility.

The Statement recognizes that hardened safe storage of facilities, such as an FRP or MOX-FFP, that contain significant quantities of long-lived radionuclide would have to be followed at some time by final decommissioning activities that would remove residual radioactivity from the site and permit final termination of the facility license. A variety of factors would determine the length of time that a specific facility would remain in hardened safe storage. The period of 1,000 years was selected as a conservative upper bound. It should also be noted that because of uncertainties surrounding the surveillance and maintenance of decommissioned facilities for long periods of time, immediate dismantlement was selected as the reference decommissioning alternative for the MOX-FFP. Dismantlement after 30 years of safe storage was selected as the reference mode for the FRP (see final Section 4.6).

DOE/ET-0028, Section 8Issue

The preliminary information offered by the DOE in Section 8.0 of the back-up document DOE/ET-0028 is obsolete and does not accurately reflect the Pacific Northwest Laboratory studies of decommissioning for the NRC as stated on page 8.1. The NRC information should be properly referenced and the DOE should provide current estimates of the TRU wastes to be expected from all decommissioning activities. (208-NRC)

WASTE MANAGEMENT OPERATIONSResponse

The decommissioning information presented in Section 8.0 of the back-up document (DOE/ET-0028) and summarized in Appendix O of the draft Statement, was based on detailed studies of decommissioning methods, costs and safety performed at PNL for NRC. The information presented is in substantial agreement with studies that have been published by NRC (NRC 1978b and NRC 1978c) since the draft Statement was issued. Information from these studies was modified as required to account for differences between the reference facility descriptions used in the NRC studies and those used in the present Statement and to account for other differences in study assumptions. It should also be noted that information such as estimated decommissioning waste volumes are presented in different ways in the NRC studies than they are in this Statement. For example, waste volumes presented in the Statement back-up document DOE/ET-00028, Vol 4., Section 8, are volumes prior to treatment and packaging. Both before and after treatment volumes are presented on pages 10.A.71 and 10.A.72 of Vol. 5. Waste volumes presented in the NRC studies are volumes of packaged waste requiring disposal. These differences make direct comparisons between the NRC studies and this Statement somewhat confusing but they are not inconsistent.

Repository Construction and Operation

Draft pp. 1.1, 1.5, 2.1.26

Issue

One commenter pointed out that the draft Statement indicates that the quantity of waste can be directly scaled to the total energy generated during the operating reactor life cycles. Thus, the 250 GWe case generated 0.64 times as much waste as the 400 GWe case. From this a reader can conclude that the number of repositories required for the alternative growth scenarios would be 1/3 less than those required for the 400 GWe scenario. However, the draft Statement does not provide information on the specific number of repositories required for the alternative scenario and this is cited as a serious omission since environmental effects vary according to the number of repositories. (43)

Response

The intent of the information presented in the draft Statement was to show that the number of waste management facilities including repositories for the alternative scenario would be 0.64 times as many as those for the 400 GWe scenario. In the final Statement, the number of repositories required for 5 different energy scenarios is presented in Table 7.3.10.

WASTE MANAGEMENT OPERATIONSDraft p. 1.4Issue

One commenter suggested using TRU waste containers as heat buffers between high-level containers in the repository. (181)

Response

It may be possible to design the repository operation so that the TRU waste containers are dispersed in the high-level waste disposal area. This might indeed provide a more efficient use of repository space but the implications of this concept have not been fully analyzed and the more conservative concept of providing separate areas for TRU waste and high-level waste was used for the conceptual repositories in this Statement.

Issue

Several letters commented on the topic of retrievability of waste from a repository.

Draft pp. 1.3 and 3.1.53--There are a few technical reasons why retrievability should be attempted and a number why it should not. Further, over any time period in which maintenance of retrievability is technically practical there is relatively little in the far-field which can be learned. The movement of radioactivity would be so slow as to produce little meaningful data. (154)

Draft pp. 3.1.105--The period of retrievability should be discussed at an earlier point in the final statement. (154)

The subject of retrievability is not treated in either a systematic or cohesive manner. More consideration should be given to the period of retrievability and the risks associated with retrievability. (58, 124)

Response

DOE agrees that little far-field data can be collected during a retrievability period of five years. However, the objective is to confirm predictions about behavior in the near-field in response to placement of the waste. It was never intended that there would be any possibility of observing movement or radioactivity which is not expected to even be possible for several hundred years after placement.

The draft Statement examined retrievability for periods of five and up to 25 years following emplacement. The final document contains a discussion of retrievability for periods up to 50 years (see Section 5.3.1.5 and Appendix K). The section on Technology Comparisons (Section 6.2) uses "Potential for Corrective Action" as one of the factors by which disposal options are examined.

WASTE MANAGEMENT OPERATIONSDraft p. 1.9Issue

The total amount of radiation in curies for each disposal site and for all sites and for each waste type should be stated. Also what percent of the radiation will still be left after 500 years? (62)

Response

The total curies of radioactivity disposed of in each of the five growth scenarios in the final Statement is identified in Tables 7.3.13 and 14. The number of repositories and thus the portion of the total radioactivity that may be disposed of in any single repository will depend on the specific site and the geologic media selected. The range of nominal 2000 acre repository requirements for each of the nuclear growth cases is shown on Table 7.3.10. Figure 1.1 in the summary shows that the amount of radioactivity remaining after 500 years will be reduced by a factor of 100 to 500 or more relative to the amount originally placed in the repository.

Draft p. 1.10Issue

One commenter requested that more detail be provided on area thermal loading limits. (40)

Response

Such information can be found in final Section 5.3, Appendix K of Volume 2, and Section 7.3 of DOE/ET-0028.

Draft p. 1.10Issue

Table 1.2 should present the repository acreage requirements for a 6,300 GWe-yr economy. (208-NRC)

Response

The relationship described on page 2.1.27 of the draft Statement shows that acreage requirements for the 6,300 GWe-yr growth scenario would be 63% of the requirements shown in Table 1.2.

WASTE MANAGEMENT OPERATIONSDraft p. 1.10Issue

One commenter requested an explanation of the method used to obtain the values in Table 1.2. (208-NRC)

Response

The basis for repository area requirements shown in Table 1.2 of the draft Statement can be found for the once-through cycle in Table 7.4.2 of DOE/ET-0028 and for the reprocessing cases in Table 7.5.3 of DOE/ET-0028. The total number of containers classified by waste type for each of the nuclear growth cases and the range of 2,000 acre repository requirements, considering the four geologic media, can be found in the final Statement in Section 7.3.

Draft p. 1.11 and Chapter 4.0Issue

Several letters questioned the range of repository availability dates (1985-2000) used in the draft Statement. (154, 181, 208-NRC, 219)

Response

The final Statement examines (under the proposed action) repository availability from 1990-2010 (see Chapter 7.0).

Issue

Several letters expressed concern with the retrievability process.

Draft p. 1.14--The retrieval capability of 5-50 years mentioned may be inadequate especially if a very serious incident occurs. (62)

Draft p. 3.1.105--The whole question of retrievability is bothersome. The statement says that "This would involve returning to the emplacement rooms, removing the spent fuel canister from its sleeved hole with the same transporter originally used for emplacement, transporting the canisters to the receiving stations where they are hoisted to the surface and providing some sort of interim storage for the canisters until another repository is ready to receive the spent fuel." This assumes that the failed fuel canister is not so damaged that it can still be handled with the same equipment used for handling one which is undamaged; that "some sort of interim storage" is available for failed fuel canisters; and, that another repository will eventually be ready. There is doubt that another repository would ever be ready should the first one not work out satisfactorily. (35)

WASTE MANAGEMENT OPERATIONSResponse

Failure of a spent fuel canister does not necessarily imply failure of the host repository. It is anticipated that providing retrievability for some initial period of time will allow for removal of waste if unexpected phenomena are observed which could lead to the failure of the repository system to provide the required isolation or containment. If retrieval is required, waste removed could be transferred to temporary surface storage. Ultimately, the waste will be disposed of in a separate repository or repackaged for re-emplacment.

Draft p. 1.18Issue

Several commenters questioned the following statement--"After return of the biota which had been displaced during construction operations, the unoccupied buffer land could provide an undisturbed wild life sanctuary." (144, 213)

Response

Except for the rock spoils pile, the land could and would most likely return to useful habitat for biota. This has been observed in many large plants. That the rock spoils pile must be dealt with is acknowledged but these form a very small portion of the site.

Draft p. 1.19Issue

Several letters noted that the statement--"Based on repository design criteria, the health effects would be zero as long as the repository performs to the design basis and no accidents occur after closure."--is overstated. (142, 154)

Response

Based on this Statement's definition of isolation, the quantities of waste which might reach the accessible environment would not be large enough to cause any health effects.

Draft pp. 2.1.3-4Issue

One commenter requested that the Statement address the repository startup and shutdown schedules (i.e., how many will be needed and on what schedule). (208-NRC)

WASTE MANAGEMENT OPERATIONSResponse

The number of 2000 acre repositories required is identified as a range of values for both the once-through and reprocessing cycle in Table 7.3.10 of the final Statement. The startup schedule will be dependent upon whether the proposed program or the alternative program is adopted and on the geologic medium selected for the initial repository, and on the extent that regional repositories are sited (i.e. how many repositories will be operating concurrently). Because of these variables, it is not feasible to identify the repository startup schedules in this Statement except to identify a range of dates for startup of the initial repository.

Draft p. 2.1.4Issue

One commenter stated that the draft did not address the question of the final disposition of very long-lived fission or activation products, such as ^{129}I , ^{59}Ni , and ^{99}Tc , which are separated from TRU or high-level wastes. Cost/benefit estimates including them with the HLW and TRU wastes should be addressed. (208-NRC)

Response

The only long-lived fission product or activation products that are assumed to be separated from TRU or high-level wastes in the assumed processing are ^{129}I and ^{14}C . The separated ^{129}I and ^{14}C is captured, immobilized, and packaged for disposal in a repository along with the packages of TRU waste. The ^{59}Ni is retained in the fuel residue (hulls and hardware) while the ^{99}Tc is retained with the rest of the fission products in the high-level waste. Thus, the effect of including these long lived nuclides with the TRU wastes has been addressed. See final Appendix A, Tables A.2.1a-A.3.9b.

Draft pp. 2.1.17, 18, 19Issue

The data on radioactivity content of the low-level TRU wastes in Section 2 of the draft Statement indicates it might not be necessary to send the low-level TRU waste to deep geologic disposal. In view of the large impact the low-level TRU waste has on repository volume, careful consideration should be given to the need for such disposal and the rationale clearly explained. (208-NRC)

Response

Although the volume of low-level TRU waste (referred to in the final Statement as contact-handled TRU waste) is quite large, it has a very small impact on repository area requirements. Table 7.5.3 in DOE/ET-0028 shows that these low-level TRU wastes utilize only 1 to 2% of the repository area. This is because they can be placed much more compactly in

WASTE MANAGEMENT OPERATIONS

the repository than other wastes that have significant heat generation or surface radiation rates. DOE agrees, however, that ultimately not all of these wastes will necessarily require disposal in the deep geologic repositories. For the purpose of this generic Statement it was considered prudent to assume that all suspect TRU wastes require geologic disposal.

Draft p. 2.1.22Issue

One commenter stated that the discussion of dynamics in Section 2.1.4.4 is incomplete. (40)

Response

An expanded discussion of system dynamics is presented in Chapter 7 of the final Statement. Additional information on dynamics can be found in Appendix A of Volume 2 and in Sections 3.1, 3.10 and 10.1-10.6 of DOE/ET-0028.

Draft p. 2.1.22Issue

The maximum spent fuel receiving rate considered for a repository in the draft Statement is 12,000 MTHM per year which amounts to a handling requirement of 10 canisters per hour. These rates appear unrealistically high and the design of the handling system to accomplish this should be presented. (208-NRC)

Response

The canistered waste handling facilities designed for this receiving rate are described in DOE/ET-0028 for receiving facilities on p. 7.4.10, for shafts on pp. 7.4.15 and 16 and for sub-surface facilities on pp. 7.4.17-20. Because lower growth rates are assumed in the final Statement, the maximum handling requirement is substantially reduced. In addition, the waste disposal requirements may be divided between two or more regional repositories.

Draft p. 3.1.35Issue

The GEIS should address retrievability in a fashion that the potential for such retrievability (over the full term of operation) can be properly addressed. Consideration of the creep behavior of salt under thermomechanical loading is important for the design of a repository in salt because it will affect the short (operational) and long (retrievability) term stability of storage rooms and access ways. (208-NRC)

WASTE MANAGEMENT OPERATIONSResponse

Conditions at specific repository sites may warrant longer (up to 50 years) periods of readily retrievable emplacement before sufficient confidence exists to permit backfilling. For a repository in salt, the amount of salt creep is also highly site dependent and may be more severe at some locations. For this generic Statement, a site was assumed where 5 years was a sufficient period of ready retrievability and salt creep was not excessive. Section 5.3.1.5 of the final Statement discusses alternative measures to allow longer ready retrievability periods.

Draft p. 3.1.36Issue

One commenter noted that seismic occurrences during repository construction and operation are the primary risk to repository integrity. (35)

Response

Due to the relatively short period of time that the repository is in the operational phase compared to the isolation phase, earthquakes during this period are unlikely. If an earthquake does occur during the construction and operation of the repository, consequences of the event are mitigated by a facility designed to resist ground motion and trained personnel and equipment available to respond to any localized destruction.

Draft p. 3.1.37Issue

The statement--"...maintaining retrievability longer than needed to reasonably assure repository operation increases the occupational and general populace risk."--is unsubstantiated. (208-NRC)

Response

The substantiation is as follows: If workers enter the repository for inspections, occupational dose will increase and simply being in a deep-geologic mine has a certain amount of risk involved. Also, the general populace risk increases because the geologic barrier is not sealed, thus decreasing the number of barriers that assure containment of the nuclides.

WASTE MANAGEMENT OPERATIONSDraft p. 3.1.40Issue

The discussion of physical protection makes sense if it refers to physical protection during the operational phase of a repository. The first paragraph on p. 3.1.41 is self-contradictory. Because operational controls will cease to exist long before any appreciable decay of ^{239}Pu , the protection must be inherent in the inaccessibility of the waste in the repository and the massive effort that would be required to remove it. (113-EPA)

Response

This discussion applies to the operational period only. Once decommissioned, the wastes are isolated from human access.

Draft p. 3.1.41Issue

The listed impacts are essentially written off without any perceived bases. For example, storage and disposal of mined mineral on the surface is a visual, as well as potential biological impact. These impacts should be fully considered and analyzed by a generic manner, and not be left for a later determination. (208-NRC)

Response

The first sentence under Operational and Post-Operational Impacts states that the issues listed need to be resolved to further clarify operational and post-operational environmental impacts associated with waste repositories in deep geologic formations; and the first of these issues is the proposed deposition of mined repository material, especially for salt repositories. On draft pp. 3.1.120-123 (final Section 5.4) there is some discussion of the toxicity of mined salt to certain plants, values given for the amount of salt deposited at the repository fence line, and comparisons of salt to other candidate geologic formations. The values for fence line salt deposition are given as 9.3 and 93 g/m^2 for the arid and reference environment, respectively, in the Statement (draft p. 3.1.121, final Section 5.4) and as 8.4 and 83 g/m^2 in DOE/ET-0029, p. 10.1.10. A similar discussion of salt impacts is given in Section 10 of DOE/ET-0029. For geologic disposal, salt was the option judged to have the potential for significant ecological impacts beyond what would result from change in land use. The acid effluents from mined pyrites from the shale formation could also be potentially damaging to aquatic ecosystems.

Issue

Several letters noted that the issue of storage and disposal of mined material was not sufficiently addressed.

WASTE MANAGEMENT OPERATIONS

Draft p. 3.1.41--Approximately 50 million tons of rock will be left on the surface during operation of the repository. This is 70 million yards of material or a mound of material 60 feet high occupying one square mile. Has the leaching consequences of this pile been addressed? Can suitable acreages be identified in the model site area to accommodate this material? How will the residual waste rock at the surface be reclaimed? (43)

Draft p. 3.1.109--There is doubt that surface storage of salt is possible for 25 years. Techniques that enable this to occur should be described in the Final EIS. (35)

Draft p. 3.1.115--The surface storage of mined material is not sufficiently evaluated as an environmental impact. A more detailed impact analysis of surface storage should be provided and cross referenced whenever it is discussed. (208-NRC)

Draft p. 3.1.120--A more detailed discussion of the ultimate disposal of excavated material is needed. In some ways this problem is analogous to the disposal of dredged material. The volumes (tens of millions of cubic yards) are similar to those involved in large dredging operations. It cannot be dismissed out of hand without more detailed discussion. (208-NRC)

Draft p. 3.1.121--What are the mitigating procedures mentioned in the third paragraph concerning salt depositions? (35)

Draft p. 3.1.121--Information should be provided regarding the size of the area over which the effects of salt dispersal would be felt. (58)

What are the plans for the tens of millions of tons of salt at the repository? (2)

What plans have been made to dispose of mined materials? (213)

Response

Sufficient site-specific information does not exist to make a detailed generic impact analysis of much value. The potential impacts beyond those associated with change in land use are those resulting from environmental release of mined salt and the creation of acid liquid leachates from the pyrites in the mined shale. Environmentally-safe management of both these mined materials is considered to be possible.

The disposal of material excavated from the repository is included in the comparison of environmental impacts (see final Section 5.4). The basis for this discussion is contained in DOE/ET-0029 (Section 4.4.1). These questions will be analyzed on a site specific basis in the EIS required for an actual repository.

Draft p. 3.1.105

Issue

The Authur D. Little work for EPA found that spent fuel heat loadings should be about the same for granite and salt. This seems reasonable considering that the salt has a

WASTE MANAGEMENT OPERATIONS

higher conductivity than the hard rocks and is surrounded by shale which is not a good conductor. We would like to correct this discrepancy between DOE and A. D. Little heat loading models. (113-EPA)

Response

Our reading of the A. D. Little report prepared for EPA (EPA 1977) indicates that the primary consideration relative to the repository heat loading in that study was the far-field effect of surface uplift. DOE's analysis did not find far-field effects to be the limiting factor for 5 to 10 year old waste except in the case of spent fuel disposal in salt. For all other cases, the thermal loading limit was set by near-field criteria, i.e., rock stresses in the emplacement region. To allow for uncertainties in the criteria and calculations, actual loadings used for the conceptual repositories were limited to two-thirds of the calculated allowable loading. All temperature profiles were recalculated and rechecked for the final Statement using an improved version of the computer code used in the draft Statement. The thermal criteria and temperature profiles can be found in Appendix K.

Draft p. 3.1.106Issue

No discussion is given of the air content of the repository following backfilling. The mobility of several significant radioelements, plutonium, neptunium, uranium, and technetium, are affected significantly by their oxidation state. (In some schemes for the in situ solution mining of uranium, air is used as the source of O₂ and is the oxidizing agent of the uranium.) The air content should be briefly discussed. (113-EPA)

Response

A room backfilled to within 0.6 m of the top will be slightly over 90% filled as rooms in the different host rocks range in height from 6.7 to 7.6 m. Since the backfill cannot be compacted to theoretical density, it is apparent that only some 70 to 80% of the room volume will be occupied by solid matter. Thus, the remaining 20 to 30% will be air. However, significant oxidation of the various radionuclides from this source of oxygen is not anticipated for several reasons.

1. Air, alone, is not expected to corrode, erode, etc., or otherwise damage the waste package.
2. Some of the air will dissolve in circulating water (ground water or brine) making ground waters somewhat more aggressive. The waste package will be designed to last 50 or more years at which time waste package temperatures will have passed their peaks and be declining. Consequently, the waste form will not be exposed simultaneously to its maximum temperature and an aggressive environment.

WASTE MANAGEMENT OPERATIONS

3. Design of the repository will insure that the maximum temperature achieved by the waste form will not be high enough for significant oxidation of heavy metal elements even if an oxidizing environment were present.

Draft p. 3.1.106Issue

One commenter was interested in what the rooms, shafts and tunnels will be backfilled with. (218-DOI)

Response

Backfill material can be previously excavated rock (see Section 5.3.1.6) or it may be special absorbant material (see Section 5.1.2).

Draft p. 3.1.107Issue

One commenter requested to know the volume of material for permanent onsite storage and whether adequate mining capability is included in the repository design. (43)

Response

Volume of permanent on site storage is included in Tables 5.3.4 and 5.3.8 of the final Statement. Adequate facilities and equipment have been included in the repository design.

Issue

Several letters commented on the placement of various types of wastes next to each other in a repository.

Draft p. 3.1.111--There should be some consideration of the possible interaction of the various wastes with each other. If the transuranic waste contains organic material, these may contain chelating materials, which could have an effect of mobilizing other waste. (113-EPA)

Draft p. 3.1.114--There can be potential problems from placing low-level wastes next to high level waste if low-level waste has significant organic constituents or other chemical incompatibilities. (218-DOI)

Response

The transuranic wastes and high-level wastes are sufficiently removed from each other in separate areas to preclude this problem.

WASTE MANAGEMENT OPERATIONS

Draft p. 3.1.113, Table 3.1.9

Issue

There is a considerable difference (in the case of salt) in the amount of storage between fuel cycles. (154)

Response

As explained in Appendix K and Sections 5.3.1.1 and 5.3.2.1 of the final Statement, the difference is due to the plutonium content of the wastes. Wastes containing increased amounts of plutonium (spent fuel and uranium-only recycle waste) have increased long-term heat generation that can adversely affect the salt formation. For this reason, the amounts of this type of waste that may be stored in a salt formation is restricted.

Draft pp. 3.1.116, 117, 120

Issue

There appears to be a contradiction between Tables 3.1.14 and 3.1.11. In Table 3.1.14 the average dust concentrations at a salt repository are higher for reprocessing waste than for spent fuel, whilst in granite, shale, and basalt, the reverse is indicated. However, from Table 3.1.11 less salt is mined per MTHM for reprocessing waste, whereas for the other three rock media, more rock is mined for reprocessing waste. In any event the concentrations of salt at the reference site are less than that in seaside air. (154)

Response

This apparent contradiction has been corrected in the final Statement (see Section 5.4).

Draft pp. 3.1.115-136

Issue

No discussion of the hydrologic design criteria of the surface facilities is given. If the site is to be designed to withstand the probable maximum flood, so state and discuss. If not, discuss the consequences of the flood more severe than the design criteria. (208-NRC)

Response

As per 10 CFR 1022, facilities are not to be built on flood plains. The base flood plain is the 100-year flood and the critical flood plain is the 500-year flood.

WASTE MANAGEMENT OPERATIONS

Draft p. 3.1.116

Issue

It appears that the resource requirements are biased in favor of salt due to poor design of repositories in other media. (208-NRC)

Response

In the absence of detailed site specific geologic data, optimization of the repository design to account for the special qualities of each medium is not possible. Instead a standardized repository design using a conventional underground layout is specified.

Draft p. 3.1.118

Issue

Table 3.1.12 presents total quantities of effluents released to the atmosphere during construction and operation of a geologic repository. The potential effects of these effluents on ecosystems should be evaluated. (208-NRC)

Response

The first sentence at the top of draft p. 3.1.119 states that "The estimated releases of the pollutants (those presented in Table 3.1.12) from construction and operation of a geologic repository would not in any case result in Federal air quality standards being exceeded at the repository boundary." It was assumed that if pollutant releases were below the limits specified in Federal standards, that no significant ecological impacts would result. Therefore, no evaluation of the potential effect of these effluents on ecosystems was made.

Draft p. 3.1.120

Issue

Little or no discussion is given of the potential hydrologic implications of repository construction and operation. For example, what would be the effects on surface drainage and downstream water quality of excavated material stored on the surface? Would the material be laid out on level surfaces, would low spots be filled in, would streams be diverted or dammed? What would happen during heavy rain and/or floods? Where would water needed for construction/operation be obtained? A description of a typical site, its construction and the hydrologic and water use impacts is needed. (208-NRC)

WASTE MANAGEMENT OPERATIONSResponse

Section 4.4.1 (DOE/ET-0029) contains a discussion on surface drainage from excavated material.

Draft p. 3.1.123Issue

It does not seem credible for water inflow through shale to be ten times that of granite. (218-DOI)

Response

The figures cited represent conservative estimates of inflow volumes during repository operation only.

Draft p. 3.1.136Issue

Justification is needed for the stated maximum surface temperature rise and uplifts. (208-NRC)

Response

Justification has been included in Appendix K of the final Statement.

Draft p. 3.1.244Issue

"Other factors influencing the time required include licensing procedures . . ." This is a really gross understatement. Taking nuclear power plants on historical precedent, the greatest delays in undertaking a project have not been in siting, designing, or other technical problems prior to construction, but in the licensing process and its procedures. This fact of life is not a condemnation of the NRC, but an indication of the difficulties in deciding technical issues relating to safety and environmental impact within an adjudicatory administrative framework. In the case of a geologic repository, these difficulties will be compounded by the present lack of established criteria and standards. (154)

Response

The sentence referred to has been deleted from the final Statement. However, it should be noted that the EPA and NRC are currently developing criteria and formulating

WASTE MANAGEMENT OPERATIONS

standards for the repository licensing procedures well in advance of the construction of the first repository.

Draft pp. A.17, 32, 46, 58

Issue

One commenter was unable to reconcile the statements of inventory of plutonium-239 in Table A.14, 29, and 42 with those in Table A.53. (6)

Response

Using a half life for ^{239}Pu of 2.44×10^4 years, one gets 16.29 grams/curie of ^{239}Pu . Multiplying this value times the curies of ^{239}Pu shown in Table A.14, A.29, and A.38, in the year 2050 (presuming the commenter intended Table A.38 rather than A.42) one gets 1790, 1840, and 27 metric tons of ^{239}Pu respectively. The first two numbers agree with the values in Table A.53 while the third value indicates a typographical error. It was reproduced in Table A.53 as 17 rather than 27.

Draft p. K.5

Issue

It is stated that 25-year retrievability required lower thermal densities. For salt and shale the decrease is a factor of two while for granite and basalt it is 2.5. Why? (208-NRC)

Response

The discrepancy was due to rounding error. The change is approximately a factor of 2 in all cases. The reason for the reduced loading (i.e. reduced stresses) was described on the same page.

Draft p. K.5

Issue

Optimization of the design for a given waste type in a particular medium would likely result in a different capacity estimates. (208-NRC)

Response

DOE agrees that this would be the case. See response to issue in this section referring to p. 3.1.116.

WASTE MANAGEMENT OPERATIONS

Draft p. K.8

Issue

Figure K.6 shows a smaller temperature increase after emplacement of waste for a repository in shale (Figure K.6, p. K.8) than for a repository in salt. (208-NRC)

Response

Figure K.6 in the draft Statement was incorrect. The final Statement contains a revised figure.

DOE/ET-0028, pp. 7.1.2 and 7.2.18

Issue

The GEIS should discuss whether retrievability in salt can be guaranteed under the expected thermal loadings. It should also discuss whether the integrity of seals in the salt repository can be maintained following closure. (208-NRC)

Response

As discussed in Appendix K of the final Statement, analyses indicate that wastes will remain readily retrievable for at least the initial 5 year period.

Details of shaft sealing techniques are highly site dependent and are not discussed in the final Statement. Discussion of shaft sealing techniques is provided in NUREG/CR-0495 (NRC 1979b).

DOE/ET-0028, p. 7.3.5

Issue

Emplacement of waste containers without the sleeves does not appear to be considered in the thermal analysis. (208-NRC)

Response

The thermal analysis considers the use of sleeves with an associated air gap surrounding the canister. This maximizes the temperature of the waste canister and was conservatively used to represent the thermal conditions of canisters without sleeves also.

WASTE MANAGEMENT OPERATIONS

DOE/ET-0028, pp. 7.4.43 and 7.5.46

Issue

The times allowed to complete licensing and construction of a repository are much shorter than those estimated by NRC. Figures 7.5.13 and 7.4.14 in DOE/ET-0028 show seven years from preliminary design to operation with one year between submission of a PSAR and construction approval. NRC estimates 10 to 12 years from preliminary design to a decision on operating approval. These longer times should be used in establishing repository availability dates as these delayed availability times may affect conclusions on the impacts of waiting until alternate methods are developed. (208-NRC)

Response

The time delays have been adjusted to reflect current estimates of the times required.

Issue

One commenter suggested that early in the impact statement discussion should appear which points out that the geologic disposal facility is to be filled with excavated material and the facility sealed in the decommissioning process. (27)

Response

Section 5.3 of the final Statement presents a description of the geologic disposal concept and those activities that will occur during the operation of a waste repository.

Issue

The GEIS should discuss the occupational hazards associated with retrieval options and acknowledge that in order to have ready retrievability all main entries, storage rooms and exhaust airways must be kept open. (208-NRC)

Response

For the 5-year period of ready retrievability assumed in the final Statement occupational hazards, both mining and radiological, are not severe. It is acknowledged that for longer periods of ready retrievability or for removal of emplaced wastes after backfill, special precautions will be necessary and difficult. See Section 5.3.1.5 and Appendix K of the final Statement for additional discussion.

Issue

Several letters noted that the Statement should address the issue of the effect of emplacing waste in a geologic repository after extended storage periods. (208-NRC, 218-DOI)

WASTE MANAGEMENT OPERATIONSResponse

Appendix K of the final Statement discusses the impact of waste age on repository capacity. DOE agrees that there appears to be an advantage in terms of repository spacing requirements for aging high-level wastes.

Issue

The rationale for the thermal and thermomechanical limits on which repository design is based is missing from the GEIS and should be provided. (208-NRC)

Response

The rationale has been added to Appendix K of the final Statement.

Issue

Different repository designs and waste storage designs should be considered for different media. (208-NRC)

Response

In the absence of detailed site specific geologic data, a standardized repository design using a conventional underground layout is specified.

FUEL CYCLESDraft pp. v and 3.1.6Issue

Several commenters suggested that discussions of fuel cycles did not emphasize that in the once-through cycle, potentially valuable nuclear fuel is treated as waste. (198, 218-DOI)

Response

The Statement compares the waste management environmental impacts of the once-through cycle with the recycle fuel cycle. Except for the waste management aspect, no attempt is made to address the overall advantages and disadvantages of reprocessing versus no reprocessing.

Draft pp. 1.11 and 2.13Issue

The established fuel cycle, deferring a decision on the ultimate disposition of spent fuel, should be addressed in the Statement. (181)

Response

The Statement does examine the implications of deferring a decision to dispose of or to reprocess spent fuel (see Section 2.1 of draft and Chapter 7.0 of final).

Draft p. 3.1.75Issue

One commenter questioned whether the statement--"The majority of nuclear wastes are residuals from defense programs."--represented a measure of volume or a measure of curies. (32)

Response

On strictly a volume basis, there is a large quantity of defense wastes on hand. However, high-level wastes from the defense program are radioactively more dilute than commercial HLW. The TRU wastes from both sources are similar. See Appendix I for additional discussion of key defense and commercial waste characteristics.

FUEL CYCLES

Draft p. 3.1.83

Issue

It is stated that--"Required activities are described for four possible fuel cycles."-- while previously it was stated that--"Three fuel cycle alternatives are considered." (218-DOI)

Response

The "four" refers to once-through, U-Pu recycle, U-only recycle (plutonium stored), and U-only recycle (plutonium disposed of with high-level waste). Only the once-through and the U-Pu recycle cycles are considered in the final Statement.

Draft p. 3.1.93, 214

Issue

Several commenters noted that the U-only recycle is not a logical fuel cycle. (6, 35, 154)

Response

Presentation of this fuel cycle has been deleted from the final Statement. Related information can be found in the support documents DOE/ET-0028 and DOE/ET-0029.

Draft pp. 3.1.98-111

Issue

Thirteen pages of description of a geologic repository for spent fuel seems out of place. (154)

Response

Consideration of the once-through fuel cycle is mandated by Federal policy.

Issue

The primary thrust of the Statement should be on the disposal of high-level wastes (U/Pu recycle fuel cycle). (154)

Response

As the present administration policy dictates a moratorium on spent fuel reprocessing, the once-through fuel cycle was one of the fuel cycle options examined in the Statement.

COSTS

Draft pp 1.10, 1.23, 3.1.133

Issue

The discussion of costs and capacities for repositories in different media for the two fuel cycles is confusing and some of the data appears contradictory. For example:

1. The unit power costs do not appear to reflect the construction costs.
2. Why does a basalt repository cost \$500 million more than one in granite if the difficulties of mining in granite are comparable?
3. Why do the construction costs between fuel cycles for the same media vary in a seemingly unrelated pattern? (208-NRC)

Response

DOE agrees that some of the cost relationships are confusing since the data necessary to obtain an understanding of the costs appear in different locations. The repository cost tables in the final Statement have been revised into a single table in Section 5.6 which consolidates all of the necessary information. This table is reproduced below for reference in answering the specific questions above. Unit power costs are shown in the last column of the table for additional reference. All repositories have the same total area (800 ha or 2000 acres).

1. The ratios of unit power costs between media differ from ratios of construction cost for the reasons below:
 - Unit power costs include the effect of repository waste capacity.
 - Unit power costs include predisposal costs.
 - Unit power costs include the effect of the cost of money on costs through discounting.

The effect of the latter two reasons on cost can be discerned by comparison of the unit heavy metal costs in column 7 with the unit power costs in column 8. The largest effect by far, however, is the difference in waste capacities (noted in column 4 in the table) in the repositories due to the different thermal loading limits in various media. Thus, while the construction cost for a basalt repository is three times that of a salt repository, the unit heavy metal cost is only 67 percent greater since 2.4 times more waste can be stored in basalt than in salt.

2. The unit costs of mining basalt were estimated to be about five percent greater than those for mining granite. Most of the cost difference, however, is due to differences in mining requirements as noted in column 3 in the table.

Waste Type	Geologic Media	Mined Quantity 10 MT	Equivalent MTHM of Waste Stored		Construction Cost millions of \$	Total Operating Cost millions of \$	Unit Cost \$/kg Hm	Unit Power Cost mills/kwh
			HLW	Other				
Spent Fuel	Salt	30	51,000		1,000	590	52	.45
	Granite	77	121,600		2,600	2,350	78	.51
	Shale	35	64,500		1,300	810	57	.46
	Basalt	90	121,600		3,100	2,390	87	.53
Fuel Reprocessing Waste	Salt	35	62,000	100,000	1,200	1,210	48	.50
	Granite	53	69,000	108,000	2,000	1,940	77	.58
	Shale	30	30,500	56,000	1,300	830	73	.59
	Basalt	59	56,000	92,000	2,300	1,740	93	.63

COSTS

COSTS

3. The construction cost differences between fuel cycles for the same repository media are mainly due to the different mining requirements noted in column 3. The rest of the differences are due to slightly higher costs for facilities and shafts in the reprocessing waste repositories.

Draft p. 1.28

Issue

A question was raised whether DOE had used a reliable contractor in developing an estimate of the cost of deep excavation into a hard rock formation. (213)

Response

The cost figures cited for the mining aspect of the disposal operation were developed by a firm recognized as an expert in development of cost information for mining engineering activities.

Draft p. 3.1.119

Issue

One commenter noted that there is an inconsistency in the unit cost of spent fuel storage stated in Table 3.1.82 and the text. (113-EPA)

Response

The inconsistency noted in the question apparently refers to the assumption that six year storage of 3/4 of the spent fuel at \$6/kg/yr and 1/4 at \$14/kg/yr should cost 36 and 84 \$/kg respectively instead of the \$30/kg and \$67/kg stated in the footnotes to Table 3.1.82.

The above calculation does not take into account the effect of the cost of money on the timing of the storage charge receipts. This effect can be taken into account by discounting the unit costs by the present worth uniform annual series factor which is 4.8 for 6 years at a 7 percent cost of money. Thus, the accurate cost calculation is $(3/4 \times 6 + 1/4 \times 14) \times 4.8 = \$39/\text{kg}$.

While draft Table 3.1.82 has been deleted from the final Statement for other reasons, similar calculations are embodied in the tables replacing it (see Section 4.9).

COSTSDraft p. 3.1.132, Table 3.1.28Issue

One commenter pointed out that for added perspective the Statement should note that the values given in Table 3.1.28 represent a range of 0.2 to 0.4 mills/kwh. (154)

Response

As noted in the tables and the text of the final Statement, the predisposal costs in Section 4.9 and the disposal costs in Section 5.6 do not include key cost elements such as cost-of-money effects and research and development costs. For this reason, DOE has elected not to state these cost segments in terms of power costs. The range of power costs for waste management are outlined in Chapter 7.0 and in the Summary and their significance relative to total power cost noted.

Issue

Several letters commented on the costs associated with accidents.

Draft p. 3.1.125 - Individual and population doses, as well as health effects, are calculated and presented in the GEIS for certain postulated accidents. The potential decontamination cost and property damage associated with the same postulated accidents should also be evaluated. (208-NRC)

Cost data do not reflect damages during storage and transportation. (14)

Response

Damages caused by accidents are generally described in terms of health effects attributable to a particular scenario. For transportation accidents, however, the requested data are available (see DOE/ET-0029).

Draft p. 3.1.133Issue

One commenter noted that the statement - "granite unit costs are less than those for shale." is inconsistent with the data presented in Table 3.1.28 on draft p. 3.1.134. (208-NRC)

Response

DOE agrees and the quotation has been deleted in the final Statement.

COSTSDraft pp 3.1.210-212Issue

One commenter noted that costs based on costs of capital of ten percent for privately-owned facilities and seven percent for federally-owned facilities are low. (154)

Response

The final Statement in Section 3.2.8.2, notes that the costs of capital employed are constant-dollar weighted-average cost-of-money rates which exclude the inflation premium. Viewed in this perspective, DOE feels that these rates are conservative (i.e. high) estimates of the actual constant-dollar cost of money.

Draft p. 3.1.211, Table 3.1.82Issue

One commenter noted that Table 3.1.82 also contains a very interesting number - the \$16.40/kg for the ⁸⁵Kr storage facility. This cost was completely ignored by EPA when it "showed" the separation of ⁸⁵Kr was "cost-beneficial"--although just barely by EPA's estimates. This cost, by itself, amounts to \$40,000 per man-rem (to the US population). In other words simply the storage cost is non-cost-beneficial by a factor of 40. EPA ought to be asked to comment on this cost. (154)

Response

The high cost of krypton separation and storage is noted in the final text in Section 4.9.2. DOE agrees that this should be taken into account in considering the cost-benefit trade-offs of krypton recovery.

Draft p. 3.1.229Issue

One commenter suggested that it should be mentioned that the relative ranking of the deferred fuel cycle alternative by total cost is significantly different depending on whether cost Table 3.1.89 or 3.1.90 is used. (113-EPA)

Response

The comments in the third and fourth paragraphs on p. 3.1.229 were intended to draw attention to the difference in costs of deferred recycle under the 0 and 7% discount cases. DOE agrees that this difference could have been more clearly emphasized. See Chapter 7.0 of the final Statement.

COSTSDraft pp. 3.1.233 and 3.1.235Issue

A commenter questioned whether 1) the Statement can estimate reporting cost to within a factor of two (+50%) and 2) if money is presently being collected from users to cover the costs of waste management. (217)

Response

Chapter 3.8 of the support document DOE/ET-0028 presents in detail the cost bases used in the Statement and discusses the effect of uncertainty on the cost assumptions. (see also final Section 3.2) In terms of commercial waste management, the only activity (involving utilities) that is presently occurring is water basin storage of spent fuel. The costs associated with this activity are assumed to be passed on to users of generated electricity.

Draft p. 2.1.244Issue

If the zero in Table 3.1.94 corresponds to the year 1986, this project was started in 1973. (43)

Response

The table is structured in such a fashion that no particular repository (or project) start-up date is implied.

Draft pp. M.55-85Issue

One commenter requested that DOE reconcile differences in various cost estimates for independent spent fuel storage facilities. (1)

Response

Estimates of construction costs for an independent spent fuel storage facility have been published in several DOE reports. Most of these recent studies use cost estimates found in the reports listed below.

<u>Title</u>	<u>Basin Size</u>	<u>Cost Estimate</u>
Economics of Water Basin Storage of Spent LWR Fuel, DPST-ISFS-78-1, Savannah River Laboratory, Aiken, SC, February 6, 1978 (SRP 1978).	5000 MTHM	\$201 Million (1978)

COSTS

<u>Title</u>	<u>Basin Size</u>	<u>Cost Estimate</u>
Conceptual Design Report for the Away From Reactor Spent Fuel Storage Facility, DOE-3547, U.S. Department of Energy, December 1978 (DOE 1978b).	5000 MTHM	\$270 Million (1978-84)
Technology for Commercial Radioactive Waste Management, DOE/ET-0028, U.S. Department of Energy, Vol. 3, May 1979.	3000 MTHM	\$200 Million (1976)

The December 1978 Conceptual Design Report is an updated estimate of the venture guidance appraisal on which the Savannah River report was based, and includes additional costs for independent service facilities. The estimates in the present Statement are based on the DOE/ET-0028 report.

The table below presents a summary comparison of costs in DOE-3547 and DOE/ET-0028.

<u>Adjustment to Common Basis</u>	<u>\$ Millions</u>	
	<u>DOE/ET-0028</u>	<u>DOE-3547</u>
Initial Cost Estimate	200	270
Add 2000 MTHM Storage Capacity	48	--
Add Service Allowance	3	--
Adjustment to Receive Older Fuel	(12)(a)	--
Include Storage Baskets	--	24
Delete 85% of Owners Cost	(46)(a)	--
Adjust to 1978 Dollar Basis	<u>33</u>	<u>(43)(a)</u>
Estimated Cost in 1978 \$ Excluding Interest During Construction	226	251

(a) Figures in parenthesis are subtracted.

Thus, the two estimates agree within 10%.

Draft Appendix NIssue

One commenter stated that bases for cost estimates for transportation of spent fuel and waste should be referenced in the final document. (113-EPA)

Response

DOE agrees. Section 3.2.8.4 of the final Statement summarizes the cost methodology for transportation and notes that additional detail can be found in DOE/ET-0028, Volume 4. The predisposal cost section in the final Statement (Section 4.9) also notes that additional detail on predisposal facility costs can be found in DOE/ET-0028, Volumes 2, 3 and 4.

COSTSDOE/ET-0028, Appendix 7AIssue

One commenter noted that the tables in Appendix 7A (DOE/ET-0028) present only the mining and construction costs for a repository. The GEIS should consider all costs that will be incurred through repository closure including operating and decommissioning costs. (208-NRC)

Response

The tables in Appendix 7A are only intended to present detailed back-up data. All pertinent costs including operating and decommissioning costs are included in the repository cost estimates as explained in sections 7.4.10 and 7.5.10 of DOE/ET-0028 and Sections 4.9 and 5.6 of the final Statement.

Issue

From the aspect of a utility regulatory commission, the Statement inadequately describes the cost of radioactive waste management. (43)

Response

The Statement presents estimates of the systems cost of various waste management options based on descriptions of conceptual facilities. The uncertainties in these cost estimates are also identified. In addition, the total cost of waste management may be found discussed in DOE/EIS-0015 (DOE 1980b) which includes a "best" estimate of the total waste management cost. The spent fuel charge data in DOE/EIS-0015 is the information that should be used by a utility regulatory commission.

Issue

One commenter requested that the Statement show range of cost estimates and how little disposal costs impact electric rates. (13)

Response

The range of cost estimates is shown in the draft Statement in Tables 3.1.89, 3.1.90 and 3.1.91 on pp. 3.1.230 and 3.1.232. The impact of disposal costs on electric rates is shown in Table 1.5 on p. 1.23 and compared to total nuclear power generation costs in the last paragraph of the text on p. 1.22.

Total waste management system costs and a comparison to total nuclear power costs are presented in Section 7.6 of this final Statement.

COSTSIssue

An issue which needs to be addressed is the expense involved for safety precautions. The technology for "near-perfect" safety can probably be found, however it will be very expensive. If it comes at the taxpayers expense this will, in effect, subsidize nuclear energy. This could make nuclear energy appear more commercially favorable than another source of energy that does not receive such preferential treatment. Or if the safety costs are born by the producer, can we be guaranteed that a private corporation can afford such a large expense on a competitive energy market. (216)

Response

The Statement develops the waste management costs (1978 dollars) for the entire reference nuclear power generating system. Costs associated with treatment, interim storage, transportation, decommissioning, and disposal in geologic repositories are presented (see Sections 4.9, 5.6 and 7.6). The Statement identifies the dollar value a consumer of nuclear power could be charged for the waste management aspect of nuclear energy production. An assumption made when developing cost estimates was that all waste management costs, whether the services are provided by private industry or by the government, will be borne by the customers of the electric energy generated by the nuclear power facility and thus are reflected as an increase in cost of power.

SAFEGUARDSDraft p. 1.23Issue

One commenter suggested that the paragraph on Safeguards is misplaced and should be highlighted as a conclusion. (124)

Response

DOE agrees that the paragraph is out of place. The structure of the final Statement has been organized to reflect this.

Draft p. 1.23Issue

One commenter suggested that the statement that regulations which are in place to protect against theft and sabotage will also be in place for waste disposal is not a reassuring one. There is no mention of the fact that in some respects it is dangerous to attempt to divert or sabotage nuclear wastes. (40, 128)

Response

The final Statement does address the hazard to would be thieves or sabatours from radiation (see Sections 4.10.1.2 and 5.7).

Draft p. 1.23Issue

One commenter noted that the statement regarding in place regulations which are to protect against theft and sabotage has not been proven to be true. Witness the theft of 245 lbs of uranium from a "Navy nuclear plant." (142)

Response

Safeguards and physical protection measures now in effect are specifically intended to protect the public from theft of nuclear material. The material in question at a navy nuclear plant cannot be accounted for. Theft is one of the possibilities and must remain so but it was also concluded that this much material did not leave the plant clandestinely. Another explanation of the missing fissile material was that it is a result of imprecise material accounting practices. It is a significant challenge to account for material where large quantities are processed in scrap recovery or waste handling procedures. The scrap and waste may be degraded process material in many different forms such as degraded solids or liquids or material impacted onto filters, or inside of pipes or inside equipment.

SAFEGUARDS

For the waste in the waste management programs the accountability problems are trivial by comparison. Only gross measurements will be required to account for bulk material and simple item count inventories will suffice to account for the balance i.e., a visual count, identification and minimum record system of each cannister, drum or container.

Draft p. 1.23

Issue

One commenter expressed concern about a statement in paragraph 2 apparently calling fission products an attractive material (for theft and sabotage) while non-proliferation strategies call for introducing fission products because of their hazardous properties. (181)

Response

This subject of attractiveness of materials for theft or sabotage is addressed in the final Statement in Section 3.2.9. A material that is an unattractive target for theft may be an attractive target for sabotage.

Draft p. 1.23

Issue

Several commenters noted that the first sentence, second paragraph, and the last sentence, third paragraph are assertions that are not backed up by analyses in this section or in later sections. They should be substantiated. Further, the second sentence, second paragraph. From a sabotage standpoint, high-level waste without plutonium might also be an attractive material and should be included in the list of material in this sentence. (208-NRC, 218-DOI)

Response

See Sections 4.10, 5.7 and 3.2.9 of the final Statement.

Draft p. 1.23

Issue

One commenter expressed a concern that the material used for the waste cannisters might also be valuable enough in the future to attract invasion of the repository (copper, stainless steel, titanium, and even gold have been mentioned...). (218-DOI)

SAFEGUARDSResponse

Materials of some value used for the cannister, are not expected to be an attractive target. The quantity of this material being used in the repository is so small compared to the amounts used and readily available elsewhere in our society that invasion of a repository for its recovery is not a credible concern. Copper and titanium and even gold have been suggested as cladding materials to provide corrosion resistance for the waste cannisters. The relatively small quantity and difficulty of recovering these materials by a mining operations makes them a very unlikely target.

Draft pp. 1.23 and 3.1.40

Issue

One commenter expressed concern regarding an inference that the safeguard provisions for spent fuel and high level waste containing plutonium must be similar to those required for facilities processing strategic quantities of plutonium. (154)

Response

The reference that spent fuel or high level waste might require the same level of safeguards as facilities handling strategic quantities of plutonium has been deleted from the final Statement.

Draft p. 3.1.40

Issue

One commenter stated that the paragraph on p. 3.1.40 under Physical Operations should be modified to correct the misconception that these materials--spent fuel or HLW from the uranium only recycle option--are desirable to steal since they contain weapons material. (124)

Response

The redrafting of the safeguards discussion in Sections 4.10 and 5.7 eliminated this concern.

Draft p. 3.1.41

Issue

Several commenters were concerned that use of armed guards, physical and procedural access controls, intrusion detection devices, secure communication systems, and contingency

SAFEGUARDS

planning for assistance in emergencies sounds like the beginning of the police state. (55,73) One of the commenter suggests "The New Tyranny" by Robert Jungk should be read to discover how Western Europe has already learned the "hard way." (55)

Response

Security measures including those enumerated in this analysis are not believed to indicate the beginnings of a police state in our society. The presence of armed guards and inspections the public accepts at every airport terminal, armed guards at many banks in our communities, armed guards on every armored truck on our cities' streets, similar protection around federal property and at army, navy, air force, marine, and other service stations are accepted without allegations of police state. Physical protection measures at a very few (certainly less than six in this country) waste isolation facilities that are located in remote areas would not contribute to a police state syndrome.

Draft p. N.6Issue

One commenter asked if an aerial radiation survey could detect a spent fuel cask that had not been breached and is located inside a building. (208-NRC)

Response

See Section 4.10.1.3 of the final Statement.

Draft p. N.6Issue

One commenter noted that the NRC has promulgated an interim rule on physical protection of spent fuel shipments (Federal Register 44, 34466, June 15, 1979). Accordingly the foot note is no longer valid. (208-NRC)

Response

See Section 4.10.1 of the final Statement where the recent NRC rule is cited.

Issue

One commenter noted that the uranium-only fuel cycle is not addressed from a safeguards standpoint although the health, safety and environmental aspects of the U-only fuel cycle are discussed. In addition, although the basic purpose of a safeguards system is identified, there is no discussion of the concepts or elements of safeguards systems potentially applicable to each waste form and storage mode. Also, the draft GEIS does not identify how much

SAFEGUARDS

effort would be needed to mine the waste nor does it address the issue of how the repository management would assure the public that all waste material is in its authorized location if faced with a blackmail threat after closure of a repository. Finally, the draft GEIS should make clear the kind of adversary that is considered when a safeguards system is designed. (208-NRC)

Response

The uranium-only cycle has been deleted from consideration in the final Statement. An expanded discussion of safeguards considerations for other waste forms is included in the final Statement (see Section 4.10, 5.7, and 3.2.9).

Issue

The possibility of deliberate reopening of a radioactive waste disposal site should be presented in the final Statement. (218-D01)

Response

Section 5.7 of the final Statement presents a discussion of safeguards and physical protection measures for geologic disposal. Following repository closure waste would be available only through re-excavation or mining. The position taken on this Statement is that theft or sabotage after closure and decommissioning is not credible, but that inadvertent intrusion must be guarded against.

GROWTH SCENARIOS

Draft pp. 1.10 and 2.1.2

Issue

Considering the 400 GWe and the 250 GWe growth scenarios described in Chapter 2 of the draft Statement, would the low growth scenario change DOE's approach to repository siting and development?

What effect would the lower growth scenario have on the selection of alternatives?
(208-NRC)

Response

In the final Statement, DOE has assumed that the maximum nuclear growth will be no more than 250 GWe in the year 2000 and it does not affect our approach to repository siting and development. The only effect of very low growth assumptions (eg., the present inventory case in Chapter 7.0) might be to eliminate the feasibility of regional repository consideration.

DOE does not see any direct relationship between nuclear growth and selection of disposal alternatives.

Issue

Several letters commented on the growth scenarios used in the Statement.

Draft p. 2.1.2--The assumption of 400 GWe installed capacity by the year 2000 is undoubtedly optimistic by at least 33%. (154)

The growth scenarios are excessively optimistic and/or do not exhibit neutrality regarding nuclear growth. (22, 35, 42, 55, 62, 63, 68)

An analysis of no growth should be included. (30)

An analysis of a high growth scenario of 550 GWe in the year 2000 and a low growth scenario of 150 GWe in the year 2000 should be included. (43)

There should be treatment of situations in which nuclear growth rises through the twenty-first century and in which nuclear generating capacity remains constant. (40)

It may be more conservative and realistic to adopt the lower estimate of 200 GWe, augmented by the estimated wastes generated by the postulated early reprocessing of U. S. and foreign spent fuel elements, defense wastes, and wastes from head end operations. (22)

Response

The final Statement quantitatively analyzes the waste management impacts of the following growth scenarios:

GROWTH SCENARIOS

1. Present inventory (equivalent to industry shutdown).
2. Present capacity to retirement (equivalent to licensing no new reactors).
3. Installed capacity of 250 GWe in year 2000 and declining to zero in year 2040.
4. Installed capacity of 250 GWe in year 2000 and continuing at 250 GWe to year 2040.
5. Installed capacity of 250 GWe in year 2000 and growing to 500 GWe in year 2040.

These scenarios do bound the range of nuclear futures which are presently thought to be reasonable. However, the data are presented in a manner which allows the reader to adjust for other nuclear industry growth patterns.

Issue

Several commenters noted that the Statement should address the issue of not generating additional waste until nuclear waste can be placed into ultimate disposal in an environmentally acceptable manner. (42, 68, 167)

Response

As noted above, the Statement analyzes a no growth situation in the nuclear industry. This analysis is conducted from the standpoint of waste management impacts and does not examine the issue of macroeconomic impacts caused by the shutdown of the nuclear industry or the environmental concerns related to the accelerated use of alternate energy sources.

Issue

Several commenters noted that the growth scenarios examined in the Statement should be tied to a discussion of the dynamics of nuclear growth vis-a-vis waste disposal. (40, 62, 113-EPA)

Response

In developing facility requirements for the Statement, DOE is cognizant of the constraints placed on the waste management system by the differing growth scenarios. The constraints have been factored into the system design (e.g., growth vis-a-vis waste disposal is presented in the support document DOE/ET-0028).

GEOLOGIC CONSIDERATIONSDraft, p. 1.3Issue

The need for in-situ testing to obtain site-specific information should be stressed.
(208-NRC)

Response

The need for site-specific information, once a site has been selected, is mentioned repeatedly (draft pp. 3.1.16, 3.1.17, 3.1.19, 3.1.23, 3.1.26 and others). Rock types, properties, and site characteristics can only be described in a general way until an actual site is selected and local investigation and testing are conducted.

Draft p. 1.6Issues

Several commenters noted that the Oklo phenomenon cannot be used as justification for geologic disposal of radioactive waste. (2, 55, 96, 97, 129, 144, 211, 213, 218-DOI) Others felt the reference to Oklo should be retained and/or Oklo should be further emphasized. (11, 13, 147, 166, 181, 198)

Response

DOE is not trying to use the description of Oklo to justify conventional geologic disposal. Oklo is, however, an example of a situation where geologic material has retarded the movement of radionuclides and contained them within the earth. The Oklo phenomena are not cited to be justification for geologic material to contain radionuclides. From investigation and measurements at Oklo, estimation of distances and rates of movement can be determined. The information obtained at Oklo is of value in expanding our understanding of transport phenomena and in helping to develop models of radionuclide transport. Oklo is meant to be supporting evidence for estimating the long-term behavior of radionuclides placed in a geologic environment.

Draft p. 1.12Issues

Why the four rock types were considered representative of all geologic media?
(208-NRC, 218-DOI)

Response

The words "of all" were deleted in the final Statement.

GEOLOGIC CONSIDERATIONSDraft p. 1.12Issue

Geologic disposal should not be restricted to salt, basalt, granite and shale. (6)

Response

DOE agrees that numerous possible geologic media exist which could be considered for disposal. However, in this Statement, only salt, basalt, granite and shale were examined in any detail. These four media are considered representative geologic media for this study.

Draft p. 1.12Issue

Two more general geologic considerations should be added to the "Six general considerations for geologic disposal":

- 7) Density and reliability of investigation required by this media.
- 8) Ease of demonstrating adequate knowledge of the host rock and its surrounding environment.

Also a sentence should be added to the last paragraph: "They (rock discontinuities) also complicate investigations of the host rock, increasing time and cost of exploration and analysis, and detracting from confidence in knowledge of rock conditions." (154)

Response

Density and reliability of investigation required by the media--a rough, general estimate could be made to characterize these factors, and some numerical or quantitative rating system developed to rank them for the four media.

Ease of demonstrating adequate knowledge of the host rock and its surrounding environment--again some rating system could be devised. For both 7) and 8), the terms "required" and "adequate" imply some sort of criteria by which the ratings would be judged.

The two suggested considerations are specific to the various media and particularly to the geologic setting in which any medium was being investigated, i.e., the setting may impose conditions on a medium that are not typical of that medium in general.

The last paragraph was amended to read "...may complicate design, and increase exploration and construction costs."

GEOLOGIC CONSIDERATIONS

Draft pp. 1.12 and 3.1.32

Issue

One commenter indicated that the media properties consider only properties which make it possible to construct a repository. (113-EPA) Several commenters noted that groundwater is discussed only during the construction phase and there appears no discussion of the likelihood of water inflow following repository closure. (113-EPA, 208-NRC)

Response

Emphasis is on the construction phase because it is believed that groundwater is more likely to enter a partially mined repository as opposed to a finished repository in a medium selected partly because of its essential lack of flowing groundwater. Groundwater transport and possible ways by which groundwater might reach a repository are considered in Section 5.5.

Draft pp. 1.12, 1.25, 3.1.28

Issue

One commenter pointed out that rock falls rather than rock bursts account for the majority of injuries and fatalities in mines. (218-DOI)

Response

The text has been changed.

Draft p. 1.13

Issue

Not only hydraulic gradients should be measured in situ but other hydraulic and geochemical parameters such as permeability, porosity, and sorptive properties should be measured on site as well. (218-DOI)

Response

The text has been changed.

Draft p. 1.13

Issue

One commenter suggested that any disposal method that requires maintenance (e.g., the need for dewatering in high permeability rock) should be avoided. (128)

GEOLOGIC CONSIDERATIONSResponse

The site selection process takes into account parameters such as the hydraulic conductivity of the rock mass. A repository would not likely be sited in highly permeable rock requiring dewatering.

Draft p. 1.13

Issue

In addition to salt's excellent thermal properties, the favorability of salt's lower variation and fewer discontinuities should be mentioned. (154)

Response

It is questionable that salt deposits as a rock group show lower variation than say masses of granite intrusions. The reader should refer to Table B.2.1 of the final Statement.

Draft p. 1.13

Issue

One commenter found no section in the draft Statement that deals specifically with site selection. The commenter felt that the Statement should note that considerable work has already been done at stages one and two and also some at stage three. (154)

Response

See draft pp. 3.1.17 and 3.1.28 under The Site Selection Process. This draft section discusses the process of site selection from the standpoint of four rock types, their properties and their occurrence. The intent is not to select a site but to describe a method for approaching the problem and to point out how the method would apply as the process leads to smaller areas and more detailed investigation.

Also, Section 2.3.1.2 of this final Statement discusses DOE's proposed site selection process.

Draft p. 1.13

Issue

In order to assess fracturing and permeability in granitic rocks, field measurements are required. Further, the commenter suggested the difficulty of properly interpreting geophysical data gathered on granitic rocks in place. (190)

GEOLOGIC CONSIDERATIONSResponse

Fracturing and permeability in granite may be difficult to detect, measure and model, particularly for a large rock mass with widely spaced or even random fracturing. See Section 5.2, which also states that field tests will be necessary after a site is selected. Geophysical methods have developed to where they are a valuable and widely used method of obtaining subsurface information. Locating and describing relatively small zones with anomalous properties (such as fractures) in a large rock mass are still difficult.

The National Waste Terminal Storage program (NWTs) recognizes that permeability tests are required in the field. During exploration for any potential repository site, permeability tests will certainly be performed.

In the Climax facility at 1400 ft. depth in Nevada Test Site granite, in-situ rock permeability measurements are part of the experimental program.

Draft p. 1.13Issue

The opening statement (under Adequacy of Data Base) has the appearance of a bid proposal and does not adequately reflect the extent of research covered elsewhere in the document. (34)

Response

This statement has been deleted.

Draft p. 1.14Issue

One commenter stated his work demonstrates the futility of laboratory measurements on rock properties for shallow granitic rock bodies (less than 1 km depth), at least in California. The commenter suggested that there remains a lot to be learned about interpretation of geophysical data gathered on granitic rocks in place. (190)

Response

The NWTs program recognizes that rock properties and behavior must be obtained in granitic rock bodies at depth. The program has participated in the Stripa tests in a mine at about 1200 ft. depth in Sweden. Analyses of data from those tests is ongoing.

At present, the NWTs program includes work in the Climax facility at about 1400 ft. depth in the Nevada Test Site. The Climax tests are just beginning and will provide field data at depth for several years to come.

GEOLOGIC CONSIDERATIONSDraft p. 1.14Issue

One commenter stated that predictions, which need be known with a high degree of certainty for only a few hundreds of years, are much easier to be comfortable with than those for 100,000 years. The commenter suggested the following addition in the second paragraph: "Some media (i.e., salt and deep sea sediments) offer opportunities for confidence in uniformity of characteristics and predictability of future conditions." (154)

Response

For the purposes of this Statement the critical period cannot be said to be only the first few hundred years--this may depend on the final waste form, type of container, type of engineered barriers, etc., as well as the disposal alternative selected. The Statement in its overall treatment at this stage considers the state of the repository and waste after several hundreds of years just as significant as before, even though the processes that can affect the repository may be different in importance than say during the high thermal activity phase for the waste.

The "uniformity of salt characteristics" comment may be misleading to readers who are not aware that most massive bedded salt contains layers/beds of anhydrite, gypsum, limestone, dolomite, shale, sandstone, siltstone, polyhalite, sylvite. (See reference 8, draft Section 3.1.1).

Draft p. 1.14, Fourth ParagraphIssue

This indicates an attempt to dismiss a real problem. Presumably water will enter any mined repository, except perhaps one above a deep water table in an arid western basin. Leachability is still an unresolved issue. (218-DOI)

Response

It is agreed that water might eventually enter a mined repository but not in significant quantities. The site investigation and testing program will characterize the repository well enough to show that sizeable amounts of water would not be present--any indication that a significant amount of water might enter the repository would likely delay or prevent use of the site.

Leachability is an area of hydrologic and waste rock interaction which is currently being studied. See final Section 5.2 and Appendix L for additional information.

GEOLOGIC CONSIDERATIONS

Draft pp. 1.14 and 3.1.11

Issue

The recent studies/work in Sweden, Canada, and the United Kingdom should be cited. The data base for granite exceeds that for basalt and shale and is rapidly approaching that of salt. Mention should be made of the sorption characteristics of granite. (218-DOI)

Response

A description of research and experience by countries other than the U.S. has been added in Section 5.2 of the final Statement.

Draft p. 1.17

Issue

In the first paragraph dealing with consequence analysis, fault offset as a primary event should be included (e.g. "faulting and rock fracturing followed by flooding..."). (154)

Response

The text has been changed.

Draft Section 3.1.1.1

Issue

The general introductory section on geology is verbose and confusing. Many of the terms are unusual or used in unusual ways. (6)

Response

This section has been extensively revised and rewritten with this comment in mind. (See Section 5.1)

Draft Section 3.1.1

Issue

There are deficiencies in the choice of references used to support the material presented. Contractor reports have not received proper peer review by the scientific community; original references used in contractor reports would have been easy to use; textbook references were old. (208-NRC)

GEOLOGIC CONSIDERATIONSResponse

Contractor reports are available at the Department of Energy's (DOE) public document reading rooms and at DOE offices, and have been since the draft Statement was released for comment in April, 1979. Reports were prepared by various subcontractors as the "Technical Support for GEIS (TM Series): Radioactive Waste Isolation in Geologic Formations" (OWI 1977, 1978) under subcontracts administered by the former Office of Waste Isolation, Nuclear Division of Union Carbide Corporation. Each report contains numerous reference lists. DOE believes it to be sufficient for the present Statement to cite these technical support documents rather than the individual references. The consultants who contributed to the draft Section 3.1 had the TM Series documents supplied to them as part of their data base, but not all the references cited in the TM Series were cited in the Statement. Several references have been added and some revisions made. For example, draft Table 3.1.2 (Chemical Composition by Oxides...) has been revised using a USGS Open-File Report (see reference number 8 in final Appendix B reference list.

Draft p. 3.1.1Issue

Salt is emphasized because this alternative has received the most study. Other media and methods should receive adequate scrutiny. (113-EPA)

Response

The Statement points out (draft Section 3.1) that the amount of available data on the various disposal techniques is unbalanced, with more data available for a bedded salt disposal method, and that more study is needed for the other media and methods. Wording has been modified from a sense of "emphasis" to one of "treated in more detail." Later revision for the final statement eliminated this comparison.

Draft p. 3.1.2Issue

One commenter felt that the four climatic factors listed to be considered in assessing the long-term isolation are not sufficient; precipitation patterns and man-induced changes must also be considered. (208-NRC)

Response

The items listed are typical climatic factors that relate to natural surficial processes as they could affect the depth requirement for a repository. Precipitation patterns, both temporal and spatial, are characteristic of a given area. Because no area is specified

GEOLOGIC CONSIDERATIONS

in these general factors, precipitation is included as part of the climatic conditions that would characterize an area when selected. Man-induced changes in the climate are not considered under the natural processes.

Draft p. 3.1.2Issue

Faulting and deformation should be mentioned in addition to erosion as hazards associated with glaciation. (208-NRC)

Response

The text has been changed (see Section B.1 of the final Statement).

Draft p. 3.1.2Issue

Several commenters indicated that predicting the lower depth of glacial erosion with any degree of certainty would be difficult. Other than uniform crustal depression, a repository located at the 500 to 600 m depth should be relatively unaffected by direct glacial processes. Future glacial front advance beyond former areas is an overly conservative assumption. (43, 208-NRC, 218-DOI)

Response

A depth of 600 to 1000 m is generally considered adequate to avoid most glacial effects. Much of the Statement is deliberately conservative in approach, particularly the accident scenarios.

Issue

One commenter made the following points regarding the role of containment and isolation:

Draft pp. 3.1.1-3--Erosion processes affect isolation rather than containment. Data sampling points may not be representative of the entire rock; younger rocks must be present in order to have confidence that the geologic history is completely known up to the present time. (154)

Draft p. 3.1.5--The statement--"Waste isolation requires that the properties of the host rock minimize transport of the waste and that the host rock be isolated from more permeable media." is true for containment and false for isolation. (154)

GEOLOGIC CONSIDERATIONS

Draft p. 3.1.14--The ratings contained in the Table on p. 3.1.14 of the draft Statement, are suspect. If they were redone, taking containment/isolation into account, the relative merits of salt and granite should increase. (154)

Draft p. 3.1.16-17--"Uncertainties" cited in the USGS Circular 779 relate only to containment and not to isolation and thus are important for only a few hundred years. "Uncertainties" should be placed in the proper perspective. (154)

Draft p. 3.1.24--Geochemical aspects of repository isolation (containment) have not yet been adequately studied. "Engineered barriers ... will probably have negligible permanence compared to lifetime of the repository," but not for the limited time period for which containment is required. (154)

Draft p. 3.1.52--If the role of containment versus isolation is properly reflected, there is no need for precise predictions of geologic events over hundreds of thousands of years. (154)

Response

The presence of younger rock is not an absolute requirement. Much geologic history has been deduced from the gaps or absences in the stratigraphic column. The Canadian Shield has pre-Cambrian rocks at the surface and is considered one of the most stable parts of North America.

The draft table on p. 3.1.14 summarizes the properties of the four media by assigning a numerical value to the property in terms of its potential for isolation (Statement usage) for each medium. All have potential for isolation of waste -- this draft table compares them by their properties. Three items have been added to the table: Plasticity, Ion Exchange Capacity and Absence of Linear Discontinuities.

Density of sampling to adequately represent the rock mass is a problem that will depend on the specific conditions at a site and can be considered after exploration and testing have begun. Also, see item 3, draft p. 3.1.51.

There is some ambiguity and perhaps confusion between the terms "isolation" and "containment." In the Statement, containment is used in a retention sense for both containers and host rock. This usage is commensurate with the definitions for "isolation" and "containment" given by the recent DOE Position Paper to the NRC rulemaking hearings on nuclear waste storage disposal (DOE 1980a).

The following definitions for isolation and containment are taken directly from this DOE Position Paper. "Isolation means segregating wastes from the accessible environment (biosphere) to the extent required to meet applicable radiological performance objectives. Containment means confining the radioactive wastes within prescribed boundaries, e.g., within a waste package."

GEOLOGIC CONSIDERATIONS

The revision of the draft includes current R&D programs that apply to uncertainties... i.e., what is being done about data gaps (see Section 5.2).

Draft p. 3.1.3Issue

One commenter suggested that, "... and sometimes earthquakes" should be added to "Environmental components important to mass-wasting processes..." (154)

Response

The text has been changed (see Section B.1 of the final Statement).

Draft p. 3.1.3Issue

For the sentence beginning "A surface environment conducive to long-term deposition ..." the comment was made that the process of long-term deposition is so slow that adding to containment is no longer of importance. Adding material at the surface would not change in any significant way, migration at depth. (154)

Response:

~~The text~~ has been changed (see Section B.1 of the final Statement).

Draft p. 3.1.3Issue

In addition to adding material on top of a repository, deposition would also lead to a change in the in-situ stresses. What effect, if any, would increased compressive stress have on the repository design? (43)

Response

The National Waste Terminal Storage program (NWTs) includes laboratory experiments, field observations, and analytical and numerical solutions to obtain information on rock properties and the behavior of rock at repository depth to the modified stress fields resulting from construction and operation of the repository. The design of the repository will include the best available information regarding the anticipated stresses and rock behavior, and a technically conservative design so that safety can be assured.

GEOLOGIC CONSIDERATIONSDraft p. 3.1.3Issue

Several commenters felt the equilibrium fringe concept is not significant for long-term isolation because of the potential for disruption of containment. (113-EPA, 124)

Response

The equilibrium fringe concept has been expanded and is only defined qualitatively---it is not a fully developed concept at this point.

Draft p. 3.1.4Issue

One commenter felt the problems listed in the draft p. 3.1.4 are considered only from a cost standpoint. Reduction of effectiveness of the repository by fracturing rock is not discussed. (113-EPA)

Response

This paragraph is part of the Canister Spacing discussion under Generic Basis for Repository Design/Waste Management Costs and is thus cost oriented. The same paragraph states that the extent of induced rock fracturing around canisters needs further investigation. Natural or existing fractures in the host rock are discussed in the same draft section with respect to rock support structures, strength, design, ground water movement, and excavation problems, as well as heat flow.

Draft p. 3.1.4Issue

One commenter recommended that the final Statement address the fact that most intense tectonic activity and virtually all volcanic activity for North America occur along global plate boundaries. (113-EPA)

Response

The text has been changed. During the site-selection process, crustal plate boundaries, areas of known active faults and zones of recent earthquakes and volcanic activity would be avoided.

GEOLOGIC CONSIDERATIONSDraft p. 3.1.4Issue

"An optimum repository will be located in a relatively stable tectonic region." One commenter requested a rationale be provided for the above quotation in the final Statement. This commenter also asked, "What is relatively stable? How is the site region defined?" (154)

Response

These points are addressed on draft pp. 3.1.20, 3.1.21, 3.1.22 for Tectonics, Seismic Considerations and Magmatism, and draft pp. 3.1.17 to 3.1.20, for The Site Selection Process.

Draft p. 3.1.5Issue

One commenter questioned the concept of "geologic stability" in the context of a nuclear waste repository which will be required to function for an extended period of time. (141)

Other commenters felt that it is not necessarily true that future tectonism may not be reflected in past tectonic history. This factor will probably have to be a probability-based estimate. (213, 218-D01)

Response

Complete and utter stability is probably not found anywhere in terms of surface erosion, deposition, and areal subsidence or uplift, particularly over hundreds of thousands of years. However, these normal dynamic processes are unlikely to affect a deep repository. The site-selection process will reject areas known to be tectonically active in favor of areas that can be shown to have been stable with regard to major tectonic activity for millions to hundreds of million years. While there is no guarantee that tectonic activity will not occur in the distant future, it can be said, based on past geologic history, that this is good evidence for continued stability.

Draft p. 3.1.5Issue

One commenter felt that faulting can do more than alter the hydrologic regime. There is no section describing disruption by faulting. (154)

GEOLOGIC CONSIDERATIONSResponse

The assumption is made in the Statement that for a deeply buried (600--plus m) geologic repository groundwater is the most likely agent to transport waste from the repository to man's environment. Over the long-term, faulting is considered as a means of forming possible conduits for ground-water flow (See "Faulting and Flooding" accident scenario, draft p. 3.1.147). For a sealed geologic repository, faulting is not considered disruptive if it does not lead to ground-water movement into or through the repository.

Draft p. 3.1.5Issue

Several commenters indicated that the definition of convection is incorrect. Convection signifies the transport of a contaminant by a moving fluid. Thermal differences may produce fluid motions and thermally-driven convection must be considered in the analysis of a radioactive waste repository. The usual driving force for ground-water flow is the head gradient. (208-NRC, 218-DOI)

Response

Convection concerns the transfer of heat by a moving fluid and can operate whether a contaminant is present or not. Convection effects as described in this section are those occurring within a repository and impermeable host rock. The repository and host rock under expected operating conditions will not behave as if they were in a saturated porous medium with flowing water. The convection effects could transport nuclides if the repository were to eventually fill with water and the heat from the waste could cause some thermally driven convection, even in the absence of any flow through the repository.

Draft p. 3.1.5Issue

Advection of nuclides with the local ground-water flow should be added to the list of major mechanisms related to nuclide transport through the disposal media. (208-NRC)

Response

Transport by migrating ground-water is not included in the general list of mechanisms related to movement of nuclides because under normal operating conditions in the essentially impermeable disposal media flowing groundwater should not occur. That is, barring any breaching of the repository that would establish flow paths, any water movement through the repository should be so slow that other processes would be more important.

GEOLOGIC CONSIDERATIONSDraft p. 3.1.5Issue

One commenter felt the section on Hydrology of the Host Rock is highly simplified; for more detailed explanations standard reference works should be cited. (208-NRC)

Response

This section is intended as an overall description of a general geologic consideration as it applies to waste isolation and site selection. Two references were added: Davis and DeWiest, and Walton (see reference list Appendix B).

Draft. pp. 3.1.5, 6, 19Issue

One area of serious concern which appears to be neglected is the effect of repository construction and operation and hydrologic barriers to long-term transport of radioactive materials from the repository. The following examples are cited:

Construction is likely to increase hydraulic conductivity of the rock mass. There is no evidence presented in GEIS to show that such factors have been considered. This is a serious defficiency.

Rock fractures, joints and fissures are potential paths for increased ground-water flow. Mine construction and testing may induce local fracture conditions that may or may not be identified in sample permeability testing. However, the in-situ extent of fractures, joints, and fissures could produce increased ground-water flow in other than direct down-gradient directions. Have such factors been considered and what conclusions have been drawn? (208-NRC)

Response

These concerns are part of Effects of Changes Induced by Emplacement of Waste Excavation (draft p. 3.1.34). In general, these effects are discussed qualitatively as concerns and sources of possible effects. Most of the effects will depend on specific site conditions which will not be known until a site is selected and some testing/excavation has been conducted.

For hard rocks, such as granite, permeability is difficult to measure in boreholes because the fractures intersecting the hole are most likely not representative of those in the larger rock mass. To obtain rock mass permeabilities a new test has been developed by Lawrence Berkeley Laboratory and is being applied in granite in the Stripa mine in Sweden. Briefly, the test involves isolating a section of mine drift, pumping in air of known humidity, and measuring the humidity of the discharged air. Also, pressures are measured in

GEOLOGIC CONSIDERATIONS

drill holes beyond the drift surface. In the experiment at Stripa, it is believed that the volume of rock on which the permeability is being measured is about 1×10^6 cubic meters.

Measurement of this type includes the increased permeability caused by mining. As yet, the experiment is not complete but we believe it to be a step forward in measurement of rock mass permeability. In addition, a few of the heater tests in granite have included borehole measurement of permeability that were taken as the rock was heated.

Draft p. 3.1.6

Issue

One commenter felt that to group geologic materials as aquifers or aquitards is misleading; a whole continuum of both permeability and porosity exists; local site conditions generally determine how one would classify a unit, since the terms are often relative. A discussion is needed of piezometric levels, and leakage between confined and unconfined units. A discussion of steady state versus transient flow conditions and the variabilities of parameters governed by the matrix plus secondary features such as faults, joints, etc., is needed. (208-NRC)

Response

Information and discussion of hydrology and ground water have been collected from draft Sections 3.1.2 and 3.1.3 (e.g. pp. 3.1.23, 3.1.32, 3.1.48) and assembled under final Section 5.1.

The discussion is intended to be an overall treatment as a consideration for site selection and to consider some potential problem areas. Potential site areas are discussed only in general terms, as are the "desirable" or favorable physical conditions. When more specific sites are designated, a more specific description with site data can be made.

Draft p. 3.1.6

Issue

Resource potential of the host rock will attract future human intrusion and increase the probability that a repository will be breached by man's activities. (113-EPA)

Response

The site selection process is designed to avoid areas of known resource potential, or what are considered resources today. If this can be done, the problem becomes one of locating a site where there are no materials, ores, etc., that could conceivably be considered resources in the future. The latter problem becomes more difficult and does involve some speculation.

GEOLOGIC CONSIDERATIONSDraft p. 3.1.6Issue

It should be pointed out that fractures and joints may also be important permeability features of shale. (218-DOI)

Response

Fractures and jointing of shale are briefly identified on p. 3.1.12 of the draft Statement.

Draft p. 3.1.7Issue

First paragraph: "...conform to geologic selection requirements..." What are these? The previously described "General Geologic Considerations" or the "Siting Criteria"? The Final EIS could be improved if the terminology were made more consistent. (154)

Response

"Requirement" refers to the "General Geologic Considerations."

Draft p. 3.1.8Issue

The confining pressure whose release causes joints should be characterized; for example, glacial retreat and thermal contraction should be named as causes of jointing in rock. (208-NRC)

Response

Thermal effects on fracturing and joints are discussed on draft p. 3.1.24 and glacial effects on draft p. 3.1.3.

Draft p. 3.1.8Issue

The way in which discontinuities can impact "the transport of waste material" should be elaborated. (154)

GEOLOGIC CONSIDERATIONSResponse

The purpose in this section is to describe the physical properties of the four specified media -- especially as they relate to isolation (or lack of it) of the waste. Ground water is assumed to be the most likely agent for waste transport, should it occur. Joints and fractures are probably the most likely candidates for flow channels, given a supply of water, the necessary gradients, etc., thus, the primary concern is with them. Thermal effects on existing joints and fractures are treated under Effects of Changes Induced by Emplacement of Waste (draft p. 3.1.34) and other places (e.g., draft p. 3.1.4). In-situ tests are specifically recommended for representative rock mass properties as opposed to laboratory test (draft p. 3.1.26).

Draft p. 3.1.8Issue

Because of the complexity and nature of deep geologic and hydrologic investigations, simple analysis using permeability, porosity and hydraulic gradients are not sufficient. Appropriate fluid and matrix parameters need to be determined; difficulties in determining them and the uncertainties should be discussed. (208-NRC)

Response

To thoroughly characterize a geologic and hydrologic setting and to model it would require a detailed knowledge of the site. This type of information would be collected during site investigation and testing. Simulation modeling would probably be used to predict performance once a site had been selected and the information could be obtained. For a generic approach to repository siting, an overall geologic and hydrologic assessment must be made with less detail. As the site selection process is described, each stage or phase will add to the detail and depth of investigation.

Draft p. 3.1.8Issue

Salt domes may deform overlying strata without penetrating them. (208-NRC)

Response

The text has been changed (see Section B.6.1 of the final Statement).

GEOLOGIC CONSIDERATIONSDraft p. 3.1.8Issue

Lower water content of salt domes compared to bedded salt should be mentioned. (113-EPA)

Response

The text has been changed (see Section B.6.1 of the final Statement).

Issue

Several comments were made regarding water migration through salt beds.

Draft p. 3.1.9--The Statement "... water incorporated in them (salt beds) was trapped when the beds were formed and does not migrate," is erroneous; fluid inclusions migrate along thermal gradients. (17, 208-NRC)

Draft pp. 3.1.9 and 3.1.35,--The assertion is made that water incorporated in salt beds when the beds were formed does not migrate. This should be qualified by reference to effects of elevated temperatures on migration of brine. Also, it should be mentioned that one of the problems with salt is that brine contained within the deposit tends to move toward heat sources, such as radioactive waste. These hot brines can be highly corrosive to some canister materials and waste forms. (17, 218-DOI)

Response

The text has been changed (see Section B.6.1 of the final Statement).

Draft p. 3.1.9, Table 3.1.1Issue

Basalt or granite do not always, or even usually, have a permeability of "nil." Shale minimum permeability is much lower than 10^{-4} , perhaps 10^{-11} or 10^{-12} ft/yr. Moreover, the key hydrogeologic parameter for evaluating these rocks as repository hosts is transmissivity, not permeability. Yet, this parameter is omitted. (30, 218-DOI)

Response

The text has been changed (see final Appendix B).

GEOLOGIC CONSIDERATIONSDraft 3.1.9Issue

One commenter indicated that the statement "Joint can be ..." is too vague. Are joints usually, often, or seldom anhydrite-filled, near vertical, unopen, moderately spaced, and generally extensive? (208-NRC)

Response

Salt, as a generic rock and mineral type can contain joints with properties as identified in the statement. To characterize the joints more specifically would require specific data at a specific location.

Draft pp. 3.1.9, 3.1.13, 3.1.14Issue

Permeabilities of granite and basalt, while low, are not nil. If they were, the repositories in granite and basalt could be located a few meters beneath the weathered layer. There seems to be no appreciation that values of permeability determined in the laboratory differ quite frequently from effective (rock mass) permeability by several orders of magnitude. (208-NRC)

Response

"Joints, fracutres and faults are generally not favorable from a geologic site selection..." See draft p. 3.1.48. These types of settings will be avoided, as possible. "Methods of evaluating free water and its movement...(for example, laboratory determinations of porosity and permeability...) but zones of fracture or joint flow are difficulat to evaluate and describe in laboratory tests." See draft p. 3.1.49. "Joints and fractures (and possibly faults) that act or could act as preferential flow paths for water can be difficult to locate and describe in terms of distribution and flow properties..." See draft, pp. 3.1.49, 50. These comments are in contrast to those made for flow in a porous medium. See fourth paragraph, draft p. 3.1.26.

Draft p. 3.1.10Issue

One commenter pointed out that in the last paragraph it is indicated that igneous rocks closely related to granite might not be suitable because of trace element and mineralogic

GEOLOGIC CONSIDERATIONS

composition. However, the "Sierra Nevada granites" are shown in the draft Figure 3.1.2. These are predominantly quartz monzonites. Would this eliminate them from consideration? (218-001)

Response

Trace element and mineralogical compositional differences between granite and closely related igneous rocks would not be the sole determining factor which would eliminate any media from consideration. In the case of the Sierra Nevada granites, long term site stability, resource potential, hydrologic regime, etc., would require detailed evaluation if a candidate site were selected in that or any other media.

Draft pp. 3.1.10-14

Issue

One commenter was not able to figure out what the maps on these paper are supposed to show. (30)

Response

The maps are intended to show location of potential repository salts, granites, basalts and shales. These maps appear in Appendix B of the final Statement.

Draft p. 3.1.10

Issue

The reference cited (#8) for Figure 3.1.1 is incorrect for this figure. The information is not found in that report. However, there is an identical map in Y/OWI/TM 36/3. This was derived from USGS Bulletin 1148. The original reference should be used especially since it is readily available to the public whereas the contractor report is not. (208-NRC)

Response

The reference has been corrected to Y/OWI/TM-36/3. See Appendix B of the final Statement. Contractor reports are available to the general public.

Draft p. 3.1.11, First Paragraph

Issue

The statement is, "Granites are basically unaltered by heat because of the high temperature of formation." Granites are subject to decrepitation at temperatures above 300°C. Cracks may also form above this temperature, leading to surface heave, followed by subsidence and cracks propagating to a water-bearing zone. (214)

GEOLOGIC CONSIDERATIONSResponse

The text has been changed.

Studies have been and are being made of the properties of granite at elevated temperatures. These studies include the Stripa tests in Sweden and the Climax test facility now being operated at the Nevada Test site.

These field studies, along with laboratory studies on small samples and development of analytical and numerical models, are being continued in the NWTS program. This program is intended to obtain sufficient information to permit design of a repository so that it will function safely for its full design life and not be impaired by such phenomena as cracks, heave, or subsidence.

Draft p. 3.1.11Issue

One commenter requested an explanation of why the areas shown in Figure 3.1.2 are favorable granitic sites; there are other areas where granitic rocks are either at or close to the surface. (208-NRC)

Response

The areas shown in draft Figure 3.1.2 are shown as potential repository granites because they are near or at the surface and, as stated, because they are large granite masses. At this stage, they are potential repository sites--there certainly are others.

Draft p. 3.1.11Issue

One commenter asked if "hard" refers to hardness (as in scratch test) or strength? (208-NRC)

Response

The statement--"Most mineral components are hard..."--refers to the resistance of a smooth surface to abrasion.

Draft p. 3.1.11Issue

One commenter indicated that in addition to thermal expansion, expansion of secondary minerals can also be caused by weathering, decomposition and alteration. (154)

GEOLOGIC CONSIDERATIONSResponse

This is true. However, weathering and decomposition are generally near surface phenomena, compared to "fresh" rock at 600-1000 meters in depth. Alteration could occur anywhere given the right conditions and causes. The effects of waste heat and stress release by excavation (draft p. 3.1.26) are believed to be of more concern.

Draft p. 3.1.11Issue

The reference cited for Fig. 3.1.2 is incorrect and it could not have been developed from the information found in Reference 9. However, it appears in Y/OWI/TM-36 and is based on a diagram in OWI-76-27. Original sources should be used. (208-NRC)

Response

The reference has been corrected to Y/OWI/TM-36/3. See Appendix B of final Statement.

Draft p. 3.1.11Issue

The final Statement should mention horizontal jointing; and other features such as veins, dikes, vugs, etc. (154)

Response

The sentence--"Joints ... to be blocky or sheet-like on a large-scale, and their orientation may be vertical and intersect at right angles and/or horizontal and subparallel to the topographic surface."--was added. "The statement--Granite masses may contain dikes, veins and occasionally fragments of other rock material."--was also added to the text.

Draft p. 3.1.12Issue

One commenter pointed out that chemical reactions also affect shales and the significance of discontinuities in shale should be discussed. (154)

Response

The text has been changed (see Section 5.1). The effects of discontinuities are discussed on draft p. 3.1.29.

GEOLOGIC CONSIDERATIONS

Draft p. 3.1.12 and DOE/ET-0028, p. 7.2.9

Issue

It would be better to use either a more recent reference than Pirsson's 1947 book or to be more selective in the data excerpted from Pirsson. For example, the silica content of the granite is rather high. It turns out that this represents a single sample from Pikes Peak (Pirsson, pg. 169). It would have been better to use Tschirwinsky's average of 90 analyses (Pirsson, pg. 169) which results in a significantly different chemical composition for an "average" granite. An alternate source of information is Clark, S.P., 1966, "Handbook of Physical Constants," Geol. Soc. of Amer. (208-NRC)

Response

The granite composition given in Table 3.1.2 has been changed to that given in USGS Open-File Report 74-158 and the reference has been changed. See Appendix B of final Statement and references 8, 11 and 23 in the Appendix B reference lists.

Draft p. 3.1.13

Issue

Figure 3.1.3 does not appear in reference 10 (draft Section 3.1.1). (208-NRC)

Response

Reference is incorrect and has been changed to reference 6 (see final Appendix B).

Draft p. 3.1.13

Issue

One commenter suggested it should be mentioned that a major problem of basalts is sedimentary interbeds. (154)

Response

This is discussed under Rock Permeability and Groundwater Flow (draft p. 3.1.32).

Draft p. 3.1.13, DOE/ET-0028, p. 7.2.12

Issue

Several commenters strongly disagree with the assertion that there is limited porosity, permeability in basalt rocks. (43, 208-NRC, 214, 218-DOI)

GEOLOGIC CONSIDERATIONSResponse

The text has been changed (see Section B.6.4 of the final Statement).

Draft p. 3.1.13Issue

The fact that basalts are layered with discontinuities, volcanic ash, "soil," sandstones, etc., between the layers was left out of the discussion of basalts and jointing. The zone between layers ranges in permeability from opened to sealed. (218-DOI)

Response

The text has been changed (see Section B.6.4 of the final Statement).

Draft p 3.1.14Issue

The selection of numerical ratings should be discussed. (208-NRC)

Response

The ratings are for comparative purposes and represent an attempt to assign a relative scale to the media properties for comparison among the media types. The intent is more to summarize rather than rank the isolation potential of the rock types. Later revision deleted this portion of the text.

Draft p. 3.1.14Issue

There should be two categories of salt, bedded salt and salt domes, so that the difference in moisture content can be emphasized. Salt domes have lower moisture content which is a major consideration.

Plasticity, ion exchange capacity, and linear discontinuities should be added to the properties for the rock types. Plasticity values would have the following relative scale values: bedded salt (3), salt domes (3), granite (1), shale (2), and basalt (1). Ion exchange capacity would have values of 1, 1, 2, 3, and 2, respectively. Linear discontinuities would have values of 3, 3, 2, 3, and 1, respectively. (113-EPA)

GEOLOGIC CONSIDERATIONS

Response

The table in question has been deleted. The reader is referred to Table B.2.1 of final Appendix B. The data cited for salt do bound the range of values one would expect for both bedded and domed salt.

The other properties requested by the commenter can be derived from those presented.

Draft p. 3.1.14

Issue

The table showing the isolation of various rock types has basalt listed as having the highest rating for the quality of permeability. This may be true on a laboratory specimen, but it is not true on a large scale. According to Meinzer (1942), of the 65 first magnitude springs in the United States, 38 come from basalt rocks. (214)

Response

When considering a sequence of basalt it is true that permeability may indeed be high as a result of sedimentary interbeds. What is intended however, is for the repository to be located in a single, thick, coherent layer of basalt rather than situated such that it intersects or crosses these interbeds. Bulk rock properties are therefore the best approximation of the characteristics of this thick rock layer. The comparison of media properties on draft p. 3.1.14 are based on laboratory testing of small rock samples which represent these bulk rock properties

Draft p. 3.1.14 and DOE/ET-0028, p. 7.2.13

Issue

Figures 3.1.4 and 7.2.4 are incorrectly referenced, are incorrect and misleading:

1. They fail to show some of the other basalt areas which should be assessed as candidates for deep geological burial of HLW, e.g., Colorado Plateau, Rio Grande Valley, San Juan Mts. of Colorado, Snake River Plains, Triassic Basins of the Carolinas, Virginia and Pennsylvania.
2. The Keweenaw Series is misplotted as is the Triassic of N.J. and Connecticut. This is not surprising as the map of the Keweenaw which was supposedly used in compiling this map (Y/OWI/TM 36/7, Figure 3-1) is illegible.
3. Reference Y/OWI/TM 36/7 is cited as a source of information for the location of the Triassic "Lavas." There is no information on the Triassic in this publication.
4. The expression Keweenaw and Triassic "Lavas" is misleading, as many of these basalts are not extrusive igneous rocks, e.g., Palisades Sill.

GEOLOGIC CONSIDERATIONS

5. Figure 7.2.4 could not have been developed from information found in Y/OWI/TM-44.
(208-NRC)

Response

The Keweenaw and Triassic locations have been corrected and are now referred to as lavas/basalts. See Appendix B of the final Statement and references 5 and 15 of the Appendix B reference list.

Draft p. 3.1.16

Issue

One commenter felt that, "General Statement" is too vague to be useful in the site selection process--it is necessary to specify how long the waste must be kept isolated and how complete the isolation must be. (113-EPA,)

Response

The text has been changed. The question of how long and how complete the isolation must be is not addressed specifically. No exact performance data are specified in this generic approach. However, in the recent DOE Position Paper to the NRC rulemaking hearings on nuclear waste storage disposal (DOE 1980a) the following National Waste Terminal Storage Program Performance Objectives are provided.

"Waste containment within the immediate vicinity of initial placement should be virtually complete during the period when radiation and thermal output are dominated by fission product decay. Any loss of containment should be a gradual process which results in a very small fractional waste inventory release rates extending over very long release times, i.e., catastrophic losses of containment should not occur."

"Disposal systems should provide reasonable assurances that wastes will be isolated from the accessible environment for a period of at least 10,000 years with no prediction of significant decreases in isolation beyond that time."

Draft p. 3.1.17

Issue

Global plate boundaries should be excluded as locations for potential repository sites.
(113-EPA)

Response

This statement was made in draft Section 3.1.31, on p. 3.1.47. The paragraph containing this statement has been moved (to Section 5.1 dealing with site selection) for emphasis in the final document.

GEOLOGIC CONSIDERATIONSDraft p. 3.1.17Issue

The impression given was that a site may be arbitrarily picked and then justified by subsequent investigation. (154)

Response

This paragraph is a continuation of topics in the preceding paragraph and should be considered with it. The wording has been changed to give the meaning intended--that non-technical factors may enter into the "selection of initial regions for investigation." Technical factors will still govern the exploration, testing and qualification of a site.

Draft pp. 3.1.17, 18Issue

Who should set the criteria relative to stages I and II of the site selection process? (218-D01)

Response

The National Waste Terminal Storage Program operated by DOE has developed criteria for geologic disposal of nuclear wastes relevant to the site selective process. These requirements are being used by DOE to guide research and development activities until formal licensing criteria have been established by the NRC. These criteria are summarized in the recent DOE Position Paper to The NRC Rulemaking Proceedings (DOE 1980a).

Draft p. 3.1.19Issue

One commenter felt that the selection process described in the draft Statement implies too simplistically that each stage is largely more detailed than the previous one. This misses the fundamental logic of a thoughtful siting process. (154)

Response

No scope definition is given for Stage I of the proposed site selection process because a large part of it can be done with available data and has been done with Figures B.6.1 through B.6.4 of this Statement. The site selection process as described is intended to proceed (given a rock type and its distribution or occurrence) from regions to areas to a site(s). The process will become more detailed in each stage, e.g., as areas are selected from within regions, and as each area of interest is examined in detail to find the next smallest areas within it.

GEOLOGIC CONSIDERATIONSDraft p. 3.1.20Issue

How important is climatic change in determining the evaluation of the hydrologic environment? (113-EPA)

Response

Climatic change is cited as ranking with tectonism and magmatism as important factors in determining the evolution of the hydrologic environment. An example is the difference in present day aquifer systems in arid climates and humid climates. This statement points out that climatic changes have helped determine present hydrologic regimes and should be considered as a factor for possible affecting future ones.

Draft p. 3.1.20Issue

It was indicated that the constraints mentioned in the last sentence are real, but possibly exaggerated and a little misleading. The Western U.S. may be more active, but at least the action/inaction can be demonstrably documented there and is more difficult to document in the east. (154)

Response

The text has been changed (see Section B.3 of the final Statement).

Draft p. 3.1.21Issue

A reference should be provided for the statement that shaking caused by earthquakes is not expected to have serious effects on the repository at depth. (208-NRC)

Response

A reference has been added (see reference list Appendix B).

Draft p. 3.1.22Issue

The commenter agrees that the statements on plate tectonic concepts being useful. The really significant theory (very old) is that the earth tends to continue those tectonic

GEOLOGIC CONSIDERATIONS

activities and patterns that have prevailed for many years; "... is correct because projection of changing stresses on a regional, not local, scale is required." This is where the plate theory provides insight. (154)

Response

The theory of plate tectonics provides a mechanism and cause for such geologic items as the distribution and occurrence of earthquakes on a world-wide basis, volcanism and regional stress patterns.

Draft pp. 3.1.22 and 3.1.47

Issue

Isotopic date boundaries and provinces may rank second to crustal plate boundaries in importance to site selection and this should be mentioned. (113-EPA)

Response

Isotopic ratio dating and province boundaries have added considerably to our knowledge of former mobile zones in the earth's crust, particularly those older than say 400 million years. However, the question of future activity along these zones is not thoroughly resolved.

Initial strontium ratios ($^{86}\text{Sr}/^{87}\text{Sr}$), for example, have been used to partly define the western limit of the North American continent (Armstrong, R. L. et al. 1977 and Peto, P. and R. L. Armstrong 1976).

At present, because of uncertain knowledge of the distribution, significance and potential for affecting a repository, it is believed premature to rank isotopic date boundaries second only to crustal plate boundaries in importance to site selection. Isotopic ratio dating has been added to the draft as an item for consideration.

Draft p. 3.1.23

Issue

If regions are to be eliminated because aquifers are present at depth, then the threshold proximity of aquifers should be defined. (154)

Response

The text has been changed (see Section B.4 of the final Statement).

GEOLOGIC CONSIDERATIONSDraft p. 3.1.23Issue

Ground water as a major potable water supply for many western states should be addressed. (208-NRC)

Response

See draft p. 3.1.48.

Draft p. 3.1.23Issue

One commenter felt it should be noted that both climate and hydrologic gradients may change with time. (113-EPA)

Response

For site selection, existing climate and hydrologic conditions describe the area at the time it is being investigated for use as a potential repository site. If these are favorable, the site selection will continue and questions of trends or changes, if evident, would be addressed.

Draft p. 3.1.23Issue

Research on thermal effects on rock permeability is all right, but should not be overstated. It may be more practical to select sites where this effect is not important. (154)

Response

Permeability is emphasized because ground-water is believed to be the most likely agent to transport waste from a repository and that anything affecting the factors related to ground-water flow should also be emphasized. Until the effects of waste heat on the host rock are more fully known, it is difficult to say where or for what rock type these effects are unimportant.

Draft p. 3.1.23Issue

The consequences of drastic changes in the surface water regime should be more carefully investigated before asserting that interior drainage is favorable. (208-NRC)

GEOLOGIC CONSIDERATIONSResponse

The potential can be evaluated when a specific interior drainage area is selected for investigation. For a first consideration the fact that all surface runoff is contained within a basin is more favorable than a basin with a perennial stream or river flowing from it.

Draft p. 3.1.24Issue

The repository host rock will not necessarily be the primary geological barrier to waste migration. Man's intrusion or natural processes may put the primary dependence on other geologic formations. (113-EPA)

Response

Site selection and qualification will consider the rates of natural processes at a prospective site and will be considered in accepting or rejecting a site. Intrusion by man's activities is an accidental or "unplanned" condition that cannot be predicted. If the estimated rates of natural processes are acceptable and if there is no intrusion by man, the host rock will provide the primary barrier to waste migration over long time periods.

Draft p. 3.1.24Issue

Thermal uplift around the repository may increase the effective hydraulic conductivities of the host rock and may even result in flow paths between overlying aquifers and the repository. (208-NRC)

Response

Heat radiated by the waste and its induced thermal stresses in the host rock and repository are described as posing the "most difficult engineering uncertainties and could have the most significant impact on the design and cost..." (draft p. 3.1.34). In general, the heat effects are expected to affect a small volume of rock compared to the volume of the host rock. Also, some consideration is given to adjusting the heat load per unit area by canister spacing or possibly reducing the waste density per canister (draft p. 3.1.35), if research and field tests show this to be necessary.

GEOLOGIC CONSIDERATIONS

Draft pp. 3.1.24, 33 and 235

Issue

Several commenters expressed concern that it may not be realistic to depend on the "self-healing behavior" (of salt) to produce an impermeable seal around the repository. The design should consider worse case behavior (i.e., the opening of thermally or mechanically induced fractures around the repository to water flow from an overlying aquifer). (208-NRC, 218-DOI)

Response

There is an impermeable boundary or sheath around the exterior of a salt dome that has existed for 20 or more million years. The need to produce an impermeable seal should only exist if construction or testing breaks this. Because of the known plastic behavior and "self-healing" properties of the salt domes at depth and the small size of a repository compared to a salt dome, it seems unlikely that fractures would propagate (from repository construction or operation) through several hundred meters of salt and remain open conduits for water transport.

Draft p. 3.1.26

Issue

How will the list of problems in the draft, with only speculative solutions be handled? (113-EPA)

Response

The problems cited are those in predicting future human activity and no specific method of handling this problem is proposed. For the accident release scenarios based on human activity, it has been assumed that the intrusion occurred. The scenario is then analyzed under the assumed conditions without assigning a probability to the human activity. For certain types of human activity, such as resource exploration, the site selection process would reduce the chances of human intrusion by selecting areas where the resource potential is as low as possible.

Draft pp. 3.1.26-28

Issue

The section on Deficiencies in Data Base is much too general and non-specific. For example, it fails to mention lack of data on long-term shaft and borehole sealing; large-scale sorption measurements; long-term verification of models; large scale dispersivity measurements; and other deficiencies. (218-DOI)

GEOLOGIC CONSIDERATIONSResponse

An effort has been made to more clearly identify the research and development needs for mined geologic disposal. See Section 5.2 and Appendix L of the final statement.

Draft p. 3.1.26Issue

The great deficiency in the hydrologic data base is actual field studies and methods for obtaining rock dispersivities and in-situ sorption studies for a variety of geologic, hydrologic and geochemical environments. (208-NRC)

Response

The Department agrees that this is an area where information is incomplete. These types of data gaps or insufficiencies are pointed out where they are known to exist and an effort is made to identify what is being done to remedy them. Both dispersivities and sorption are currently being studied. See Section 5.2 of the final version.

Draft p. 3.1.27Issue

If aqueous radioactive waste is that which has been leached from the solid form by groundwater, this should be said. Otherwise people might think there is a plan afoot to bury liquid wastes. (181)

Response

The statement has been deleted.

Draft p. 3.1.28Issue

The discussion of the design of the repository considers most of the factors which would influence the isolation of the waste. However, there is no consideration of the possibility of underground collapse. The literature is filled with accounts of underground mines which have collapsed, and caused progressive fracturing to extend upwards toward the surface. Some of these failed mines are several hundred feet deep. None of them are more than a few hundred years old. (214)

GEOLOGIC CONSIDERATIONSResponse

Collapse of tunnels, cavern walls and roofs and the engineered methods of counteracting and preventing this is discussed in Section 3.1.2. Considerable experience from the mining and construction industries indicates that during the operational and retrievable storage phases this should not be a problem. Once a repository is backfilled and sealed, collapse or partial collapse into the remaining open space in the future is a possibility. This is not considered to be a threat to the repository integrity and could eventually be beneficial by adding additional rock material or increasing the density of the backfill in the repository.

Draft p. 3.1.29

Issue

Salt and abyssal-sea clay are more favorable than other media with respect to "unforeseen rock conditions," "number and spacing of fractures" and assurance that significant discontinuities are not overlooked. (154)

Response

This is true. However, these properties are only part of the total conditions and properties for a site/disposal medium.

Draft pp. 3.1.30 and 3.1.116

Issue

One commenter pointed out that ground support in a shale repository at a depth of 600 m is likely to be a major and costly problem; support costs on draft, p. 3.1.116 are clearly understated. (208-NRC)

Response

See draft, p. 3.1.31, first paragraph. It is pointed out that shales are the most difficult to support in underground openings; that tunneling and support could add significantly to the costs; that a study at the Nevada Test Site has concluded that costs in shale could increase at least 25%, and could be much greater.

Draft pp. 3.1.30-31

Issue

The draft Statement missed stressing its main point because of attempts to discuss minor problems. For example, the positive statement "...strength has major impacts on

GEOLOGIC CONSIDERATIONS

design and cost; but in all rock types, engineered support can be designed using current techniques" almost gets lost in the discussion of problems. (154)

Response

In rewriting this section (Section B.2 of the final Statement) attempts were made to place equal emphasis on the problems as well as the engineering techniques which might be used to remedy these problems.

Draft p. 3.1.31Issue

Since closure rates are expected to be high, the GEIS should describe the support systems and expected closure rates and the effectiveness of the support systems. (208-NRC)

Response

The support required for main corridors, crusher rooms, site conditions will have to be based upon 1.) how long the repository is operated retrievably and 2.) closure rates of the repository medium. The design will probably not be for a permanently open facility like a metropolitan transport tunnel, but designed for a certain operating period, possibly until the repository is filled and sealed. The design will be based on the physical properties of the particular medium at a particular site, and won't be final until these specific properties are known.

Draft p. 3.1.32Issue

The draft Statement did not reflect fissure and joint permeability differences, and induced characteristics due to construction. The draft Statement makes an assertion that is not supported; i.e., no bases have been provided to support the conclusion that groundwater inflow can always "...be controlled by state-of-the-art engineering technology." (208-NRC)

Response

This topic, under the "Generic Basis for Repository Design/Waste Management Costs" Section, does not distinguish between "joint and fissure permeability" and that induced by construction. Repository design will be influenced by data gained through the site selection process and the testing of a site before construction begins. Site testing will give some idea of the extent of fractures and joints, and state-of-the-art technology can make some estimates of the effects of these features as related to construction. Presumably, site

GEOLOGIC CONSIDERATIONS

selection and testing would disqualify any site which had joint-fracture characteristics that could lead to severe problems or uncontrollable water flows.

Draft p. 3.1.32Issue

The questions of shaft and borehole plugging apply to all media, not just salt. (218-DOI)

Response

This section of draft was revised. Questions associated with borehole sealing and research currently under way on these questions is discussed in the Final Statement in final Section 5.2.3.3 and in final Appendix L.

Draft p. 3.1.32Issue

This section about Rock Permeability and Ground-water Flow accurately describes the problems of ground-water flow, particularly through fractures. The statement that "Ground-water flow into repository shafts and rooms can be controlled by state of the art engineering technology", is only true of a mine which operates wet. The statement is not true if radioactive waste is to be isolated. (214)

Response

For repositories located in formations other than salt, the assumption is made that the repository will fill with water after closure. The capacity of the repository to isolate waste is then a function of the ground water flow rate, the transport rate of radionuclides through the formation, the leaching rate of the waste form, and the decay rate of the radionuclides. Because many uncertainties are involved in calculating the migration of radionuclides (arising in part from uncertainty in the validity of the models and parameters used in them) an additional level of protection will be added. This level of protection will be provided by the engineered barriers that make up the waste package. These barriers include the canister, the overpack sorption materials and backfill. The package itself will be designed to contain the waste under repository conditions for hundreds of years which allows for decay of a majority of short lived radionuclides with a corresponding reduction of waste temperature. The reduction in temperature after the life of the package is reached reduces the leach rate of the waste form and thus the concentration of radionuclides available for migration in ground water.

GEOLOGIC CONSIDERATIONS

Draft pp. 3.1.33 and 3.1.138

Issue

One commenter questioned why 600 m is selected as the disposal depth. (40)

Response

The use of 600 m is an arbitrary depth selected from work done at the Carlsbad, New Mexico site. It is based on the depth (700 m) and the thickness (600 m) of the Salado formation. See Section B.1 of the final Statement.

Draft p. 3.1.33

Issue

Ground-water flowing into operating mines (in Canadian Shield Granite) is probably evaporated by ventilation airflow. In the long run, seepage rates low enough to appear negligible by visual inspection are expected to be significant. (208-NRC)

Response

See draft p. 3.1.,49, where it is mentioned that flow rates and velocities that are insignificant over a 50-year period could be important over hundreds to thousands of years, and that it is reasonable to assume as one possibility that free water, over a period of thousands of years, may enter the repository (with the possible exception of a salt repository). It may be unrealistic to believe that any repository (excepting salt) would remain totally free of water for times up to a million years or longer. However, with careful site selection investigation, construction and a detailed knowledge of the site it may be possible to show that the time of water ingress and/or the rate are such that the repository serves its purpose.

Draft p. 3.1.33

Issue

This section states "Thus, cost considerations dictate that the depth of emplacement should be minimized, whereas isolation requires that the depth be maximized." The first part of that statement is sufficiently clear. However, it is not clear that the second part of the statement is correct or if correct, significant. The support for this part of the statement is qualitative and intuitive rather than quantitative and rigorous.

Geological Survey Circular 779 states: "The suggestion of Winograd (1974) that waste be placed at relatively shallow depths (30 to several hundred meters) in the thick (as thick as 600 m) unsaturated zones of the arid Western United States deserves consideration." We concur.

GEOLOGIC CONSIDERATIONS

The Teknekron, Inc. report prepared for PNL, "A Cost Optimization Study for Geologic Isolation of Radioactive Wastes," May 1979, does not indicate any significant advantages to great depths of burial except the reduced probability of repository disruption. If the large meteorite strike is truly improbable and if erosion and glaciation can be avoided (at least during the first 10's of thousands of years) then there may not be any advantages to great burial depths, only disadvantages.

The following questions should be addressed:

1. Are there regions of the U.S. otherwise suitable for a repository which can provide a safe environment for the waste at relatively shallow depths without a meaningful threat of interruption by natural events?
2. If so, what is the reduction of risk between such a repository and a deep repository (and what is the increase in cost)? What is the potential for an increase in confidence which could result in a more complete site characterization and simpler modeling of a shallow versus deep repository?
3. If not, what is the quantitative reduction in risk as a function of depth for a deep repository? (208-NRC)

Response

For the reconnaissance and generic treatment of the geologic considerations, containment of the waste and isolation from man's environment were the prime considerations. Depth to the repository has been an unresolved issue (draft p. 3.1.25) and the early estimates of depth were made for bedded salt (draft p. 3.1.48). In general the reduced probability of disruption provided by deeper burial was assumed to be desirable. Because neither a rock type or geographic area is specified in this Statement and because cost was not considered to be an environmental factor, cost-depth considerations were only treated qualitatively.

The Winograd concept would store waste in mesas and buttes in the arid west. The waste would be placed in unsaturated (above the water table) material and could require shallow depth because of the high elevation of the unsaturated material above the surrounding land surface and/or the relatively great depth to saturated material. However, mesas and buttes are erosional remnants that are exposed to surficial processes and they are being reduced in area even under today's arid climate. These features were considered more temporary than a buried rock unit that is not exposed to surface processes. The concept is worth consideration as a special case and could well be useful if a time of containment is firmly fixed and if the present arid conditions remain unchanged.

Sections 5.1 and 5.2 of the final Statement discusses disposal of waste, in geologic units within the earth, from a generic standpoint; depth of repository is one of the relevant geologic factors mentioned and discussed, but in general terms rather than specific. These factors all are concerned with the location and/or performance of a repository. The

GEOLOGIC CONSIDERATIONS

questions above are beyond the scope of the generic treatment in this Statement, particularly because cost was not considered an environmental factor and because there are no specific criteria for depth, length of time necessary for containment/isolation, or even what repository behavior can be considered isolation.

Draft p. 3.1.34Issue

The comment was made that the effect of mining on rock depends on rock type and mining technique, fracturing can be controlled by existing techniques. The greatest changes occur within a relatively short time following excavation, and that stress relief in most rock types decreases dramatically following emplacement of supports. (154)

Response

The text has been changed to read "It should be noted that the greatest change in the rock occurs within a relatively short time following excavation." See also draft pp. 3.1.30-3.1.31 under Rock Strength and Excavation Stability.

Draft pp. 3.1.34-36Issue

One comment was that the discussion of the effects of heat in the waste on repository design and construction section is not crisp. This section should recognize that if a realistic position is taken as to cooling time prior to disposal, the heat effects will be significantly reduced and that they can be further reduced if desired. (154)

Response

The draft Statement discusses a variety of waste forms and predisposal options, and assumes certain conditions for the purposes of analysis. This is intended to cover a variety of options and the cooling time is one of the factors about which a final decision has not been made. The Statement does mention decreasing the heat output per unit area by spacing between the canisters and by decreasing the waste density per canister.

Draft p. 3.1.35Issue

In the last paragraph, the resins used with resin grouted bolts are polyester, not epoxy. (218-DO1)

GEOLOGIC CONSIDERATIONSResponse

The text was changed.

Draft p. 3.1.36Issue

The basis for the conclusion that costs for additional support necessitated by the reduction in rock strength caused by radiation are not expected to be significant should be given. (208-NRC)

Response

Some data are presented in paragraph three for salt. This paragraph points out that adequate knowledge of radiation on rock properties is lacking. Based on the information available, it was concluded that the known effects on costs would be small compared to the total costs. An additional reference was added to the list at the end of Section 5.2 (reference 26).

Draft p. 3.1.36Issue

Regarding seismic loads, a brief description of how underground structures respond characteristically to earthquakes would be appropriate. (154)

Response

A reference has been added for seismic effects on underground structures under Tectonic Considerations Seismicity and Magmatism.

Draft p. 3.1.41Issue

Operational difficulties which may prevent sealing are not discussed; it may be difficult to backfill or retrieve if a repository becomes flooded. (208-NRC)

Response

The site investigation and testing program should characterize the repository well enough to show that there would not be problems like flooding--any indication of these kinds of problems would likely delay or prevent use of the repository.

GEOLOGIC CONSIDERATIONSDraft pp. 3.1.47-53Issue

The discussion in this section repeats that of Section 3.1.2. (154)

Response

Sections 3.1.1, 3.1.2 and 3.1.3 have been reorganized to eliminate repetition.

Draft pp. 3.1.47-76Issue

Discussion of brine migration is missing; no mention is made that sorption by salt is different than for other media. (208-NRC)

Response

Mention of brine inclusions in salt migrating toward a heat source was added (with reference) to draft, p. 3.1.9, and the table of physical properties comparisons (draft, p. 3.1.14) has "Ion Exchange Capacity" added, with salt given the lowest rating. Sorption is discussed briefly on draft p. 3.1.5, and listed under "Deficiencies in the Data Base" (draft p. 3.1.26) as a significant item.

Draft p. 3.1.47Issue

An additional premise should be added to the effect that certain geologic conditions will be avoided and others can be coped with through mitigating measures. (154)

Response

The two premises listed are basic to the concept of geologic disposal and are the reasons for considering geologic disposal. They apply equally to all the media discussed. To distinguish between conditions that would lead to avoidance or to acceptance with mitigating measures is more specific to each medium (i.e., not as general) and was not included here.

Draft p. 3.1.47Issue

Hydrology should also be a prime consideration in geologic site selection. (218-DOI)

GEOLOGIC CONSIDERATIONSResponse

Hydrology is discussed as an important factor in selecting the site, qualification of the site and assessing the potential for retrievability of the waste. The subject is divided into surface hydrology (draft p. 3.1.47) and groundwater, as a separate topic, (draft p. 3.1.48-49). The final Statement includes Hydrologic Considerations under Factors Relevant to Geologic Disposal (See final Appendix B), and then under final Section 5.1.1.2, as one of the most important factors in repository site selection.

Draft p. 3.1.47Issue

One commenter pointed out that for tectonics and seismicity, the critical point to demonstrate is that the activity pattern is such that future action is not likely to occur during the repository's life. This should be expressed in terms of probability and acceptably low probabilities need to be defined. (154)

Response

This is true and hopefully will be done. The difficulty is in assigning acceptable probabilities to events in the distant future, as well as defining what is acceptable in terms of these probabilities. One solution is suggested in the second paragraph draft p. 3.1.53. This problem is receiving study.

Draft p. 3.1.48Issue

The emplacement medium should be of an older age, and be overlain and surrounded by rocks of younger age, so that the absence of significantly adverse activity for an adequately long time period can be demonstrated. (154)

Response

The requirement that the medium be overlain and surrounded by younger rocks may be unnecessarily restrictive. For a disposal medium and site location, the stability and character are more important than whether or not there are younger rocks overlying the disposal material. See the comment for pp. 3.1.1-3.1.3 about the Canadian Shield, for example.

Draft p. 3.1.48Issue

Regarding jointing, faulting and fracturing, one commenter recommended the addition of the following sentence: "They increase the time and cost of investigations, complicate the

GEOLOGIC CONSIDERATIONS

representative quantitative modeling necessary for design and decrease confidence that all conditions are known." (154)

Response

The text has been changed (see Section B.2 of the final Statement).

Draft p. 3.1.48Issue

One commenter indicated that the first statement under ground-water implies that a repository can be sited in conjunction with a useful ground-water source and not affect its quality. (113-EPA)

Response

The text has been changed. The intent of the statement is not clear from the wording. The central idea in discussing ground water as resource is that it must be preserved.

Draft pp. 3.1.48, 49, 64Issue

Table 3.1.49 of the draft states that there could be an unacceptable 50 year body dose as a result of ground-water transport of radionuclides by the year 2050; pp. 3.1.48 and 3.1.49 state that insignificant flow rates over the short term may be a problem over the long term. (208-NRC)

Response

Table 3.1.49 of the draft presents the results of a "faulting and flooding" accident scenario that "assumes an improbable combination of events..." (p. draft 3.1.148); this is only true if all the given assumptions are made--a fault breaches the repository, 100 cubic ft/sec of water flows into the repository, past the waste, and enters man's environment. The other statements refer to an intact repository and not to the special "accident" scenario.

Draft p. 3.1.49Issue

Several letters noted that the statement--"Flow rates and velocities of groundwater that are insignificant over a 50 year period will have to be considered over hundreds of thousands of years." does not make sense over the long-term. (113-EPA, 154)

GEOLOGIC CONSIDERATIONSResponse

This Statement is intended to point out that, over long time periods, slow or low-rate processes could have effects that are not obvious over a short time period, and should be considered.

Draft p. 3.1.49

Issue

The chemical nature of any aquifers around a repository should be briefly discussed, including oxidation-reduction considerations. (113-EPA)

Response

Aquifers are discussed from a generic standpoint and without any specific location specified. The chemistry of the ground-water is not specified either, because the chemical nature of the water will vary depending on the location of the site. The solubility and sorption effects on the various ions and elements will also depend on the aquifer material characteristics as well as the waste form. Because of the-site specific nature of the ground-water chemistry and the effects that even small differences in water chemistry could have, this question can be better addressed when water analyses for a particular setting are available.

Draft p. 3.1.51

Issue

The statements that some issues may not be resolved with the necessary degree of certainty seems to conflict with the very next sentence, which states that uncertainties can be reduced to acceptable levels. (208-NRC)

Response

The intent is that more research may be required to reduce the degree of uncertainty to acceptable levels.

Draft p. 3.1.51

Issue

One commenter was concerned about the following statement: "...acceptability criteria may need to be adapted or modified, or even developed if unexpected conditions are met." It is important that the implication not be left that justification for a particular site can be an ex post facto exercise not based on scientific and technical grounds. The rationale and bases for criteria development and application need to be elaborated. (154)

GEOLOGIC CONSIDERATIONSResponse

There are presently no quantitative criteria for repository or site qualification. Until there are and until some site specific data are available, it is not possible to say what the acceptability criteria are. These may be proposed and then need to be adapted or modified as the site testing is conducted. At this time, the primary objective is to establish criteria that protect the environment and the public safety, both for the short and long-term.

Draft p. 3.1.52Issue

One commenter felt the belief that all problems can be solved by major efforts is unjustified. Investigations into a basic research area do not necessarily have satisfactory outcomes. (113-EPA)

Response

The text has been changed (see Section 5.2). The Statement does not say that the problem would be solved, but that a serious research effort would yield results. The capability may never be developed to predict exactly what geologic phenomena will have occurred at a given location by the end of another million years. Research into the processes that affect and change the earth would certainly yield information on the processes themselves, their interactions, the driving forces, rates of activity, timing of events, etc.. All of these items represent areas in which full understanding is lacking. As more understanding is gained, some of the basic uncertainty in predicting geologic events could be reduced or removed.

Draft p. 3.1.53Issue

Concerning long-term geologic stability of a geologic repository, the statement, "It (long-term geologic stability) only becomes a concern if and when the test facility or conservatively-loaded facility becomes a full-scale repository, and only then when the period of retrievability has ended and the repository is sealed." This statement is true, but the long term stability certainly must be considered in depth before the repository is sealed. (214)

Response

Long-term stability of a repository is a major consideration of the NWTS program. Laboratory and field observations of the behavior with time of geologic materials are being

GEOLOGIC CONSIDERATIONS

made. The results of these ongoing studies will be considered in the final design of the repository system.

Draft p. 3.1.54Issue

The statement that for longer-lived nuclides a delay in release to groundwater will probably be provided by the host geology, is not true of salt since it does not absorb great amounts of the nuclides that leach out. (62)

Response

The text has been changed.

Draft p. 3.1.67Issue

The current geologic estimate of the age of the earth is 5 billion years, not 10 billion as given. (208-NRC)

Response

The text has been changed.

Draft p. 3.1.68Issue

Lithosphere/Biosphere Transport: While the reader is correctly advised that "Some ground-water and transport models have been calibrated," he is not told that modeling of flow through fractured aquifers is in its infancy. (218-DOI)

Response

DOE agrees with the commenter and notes that these points are well taken. Final Section 3.4.3.2 discusses the limitations of mathematical models. Appendix L of the final Statement identifies research currently underway which is designed to better the understanding of the nature of fracture flow, thereby, aiding in the development of more accurate fracture flow models.

GEOLOGIC CONSIDERATIONSDraft P. 3.1.71Issue

Sorption of radionuclides is controlled by the site-specific geology. It seems unlikely that radionuclide behavior data from one site can be applied to another site. (113-EPA)

Response

Radionuclide behavior data from one site cannot be applied to another site with precision; however, they appear adequate for use in a generic statement.

Draft p. 3.1.104Issue

A major deficiency in the design of the repositories in granite, shale, and basalt is that they have been designed as if the host rock were salt. The repositories in the four geologic media should not be of similar design. For instance, the inherent structural characteristics of granite have not been taken into consideration. The design of a mine in hard rock is substantially different from that in salt. Where, by the nature of the material, a repository in salt is confined to a single level, a repository in massive granite need not be. The long term stability of large rooms in granite is well known. (208-NRC)

Response

Room and pillar mining provides for efficient use of the rock formation for the emplacement of nuclear wastes. The concept of room and pillar mining was tailored to the rock structural characteristics of the four rock types shale, salt, basalt and granite as plainly indicated in Table 7.4.2. The multiple level concept for room and pillar mining in repositories is constrained only by the physical dimensions of the geology. No technical basis is known to disavow multiple levels. In the case of bedded salt, multiple levels may be physically impossible since it is a horizontally layered geology. Room height may be of the same order dimensionally as the geologic layer thickness, thus restricting a repository to one-level thickness does not entirely rule out multiple levels though. With a thinly layered geology multiple levels may be possible if multiple thin layers exist as in the case of the WIPP site.

Domal salt has a potential for providing for multilevel emplacement since its geologic vertical dimension is a couple magnitudes of order greater compared to the room height dimension. Therefore salt repository design is not confined to a single level.

Rooms in granite may be extremely stable. Thermal/mechanical analysis for the pre-conceptual design stages has been performed. Thermal/mechanical models are presently being developed to address the impact of the heat generated by nuclear wastes upon room stability in granite repositories.

GEOLOGIC CONSIDERATIONSDraft p. 3.1.124Issue

What the GEIS is really discussing is the creation of flowpaths by creating fractures or opening fractures that already exist. The question, then, is how does the predictive model treat the flow of liquids and transport of dissolved radionuclides through fractures? Both flow and transport could be significantly different in fractures than in porous media. We know that retardation is less and, also, that retardation is the most important attenuation mechanism that has been modeled. The term "thermally-induced permeability" does not convey the difference between porous flow and fracture flow. (208-NRC)

Response

The predictive models used do not treat flow or transport through fractures. A prevailing assumption for siting any nuclear waste repository is that ground-water flow rates will be very slow. If the flow velocity is high, then the repository will be located elsewhere.

Retardation is not the "most" important attenuation mechanism modeled. Several dispersion mechanisms exist that have a greater effect on maximum peak values. Examples include leach rate and spacial distribution of the waste in a repository. In certain instances, decay rate may have a larger attenuation than retardation.

Draft pp. 3.1.136 and 3.3.3Issue

Areas of uncertainty common to different alternatives should be treated equally. Technology for long-term sealing, which has not been demonstrated, also does not receive uniform evaluation. (208-NRC)

Response

This type of treatment is being attempted in the revised Statement. Some of the problems are not the kind that past experience has dealt with and others that may appear similar among the alternatives really are not. For example, to date, no structures have required a design life of even say one-thousand years, let alone to 10 thousand years. It is impractical to try to demonstrate a thousand-year proof period in real time. An example of apparent similarity is the sealing of a very deep hole and a mined repository in a deep hole, the "repository" is part of the drilled hole but in a mined repository the hole or shaft is only the access; the repository is constructed away from the hole and can be sealed separately. However, sealing the drilled surface to repository level will present similar problems.

GEOLOGIC CONSIDERATIONSDraft p. 3.1.148Issue

First paragraph, last sentence reads, "It is doubtful that any fault would form a continuously permeable conduit to the repository, even if a fault should occur through the repository to the land surface." Faults often form conduits which transport water, and are often located by the presence of springs along their length. The study by Meinzer (Meinzer 1942) states that 3 of 65 first magnitude springs in the United States are located in sandstone, and that they are believed to owe their existence to faults or other special features. (214).

Response

The influence of postclosure faulting on the capacity of a repository to isolate waste will be considered in the analysis of repository performance. Currently the modeling capability exists to conduct detailed calculations of the effects of post-closure faulting on radionuclide migration. Preliminary calculations have been done for a hypothetical repository in salt and are published in a document entitled Test Case Release Consequence Analysis for a Spent Fuel Repository in Bedded Salt (Raymond et. al. 1980). These calculations assumed a vertical fault through the repository and overlying aquifers and extended to the ground surface. Because of the nature of the test case, many of the parameters had to be assumed; therefore, a sensitivity analysis was conducted to determine how changes in parameter values affected the resulting dose-to-man calculations. These results are published in Paradox Basin Sensitivity Analysis (Bond and Kaszeta 1979). These types of calculations will be done for all repository sites to determine the effect of post-closure faulting even though the likelihood of faulting is small.

Draft pp. 3.1.228, 3.3.22, 3.3.27, 3.3.30Issue

Maintenance of the integrity of shaft seals, room seals and canister seals (particularly in salt) would be expected to pose greater problems than in Very Deep Hole disposal. (208-NRC)

Response

The entire question of sealing is still under study (See Section 5.2 of final Statement). For the underground mined repository, it has been proposed to backfill the rooms with material removed from the repository during mining, thus completely surrounding the waste with host rock material. The shaft(s) to the repository are a more complicated problem because of the various rock materials between the repository and the land surface. The very deep hole is similar to the mined shaft in this respect plus having drilling mud

GEOLOGIC CONSIDERATIONS

effects and casing (pipe), if used, and being more directly "connected" to the waste location. Salt, depending on the depth, is expected to deform plastically and may eventually seal the backfilled rooms to original density material. The shafts into the salt unit are expected to be the main sealing problem.

Draft, p. 3.1.237

Issue

The questions posed in the first paragraph are correct, but this does not mean that boundary conditions/characteristics cannot be defined now as minimum criteria to be applied at least during siting.

It is true that standard techniques for nondestruction analysis of geologic formations are not complete nor generally available. However, this varies with the media and its environment, and with the type of information desired. For example, qualitative high-resolution data on the nature and distribution of abyssal sea clays and their contained discontinuities can be easily obtained by seismic reflection profiling. Quantitative data, however, are not yet so readily available using such techniques as they are for sedimentary environments on land. For granites and many basalt environments, seismic reflection profiling is minimally useful. The point that should be made here is that an optimum set of destructive and non-destructive exploration techniques will be applied on a site by site basis with recognition of the resulting trade-offs between accurate knowledge and troublesome penetrations. The possible notion that progress should await the development of new techniques should be dispelled. (See p. 3.1.240, 5th paragraph, next to last sentence.)
(154)

Response

During preparation of the final Statement this section was revised. Section 5.2 of the final Statement discusses the R & D needs to enhance existing site characterization analysis techniques. Both standard geophysical and non-destructive techniques are discussed in this section. Additional information regarding R & D needs and programs to meet these needs can be found in the final Appendix L.2.

Draft p.3.1.237

Issue

There is no discussion of research needs in the hydrologic transport aspects of geologic disposal. Of prime importance are the chemical and thermal interactions involving dissolved wastes and the natural rock. (113-EPA)

GEOLOGIC CONSIDERATIONSResponse

Such discussions appear in Sections 5.1.2 and 5.2, Appendix B.4, and Appendix L of the final Statement.

Draft p. 3.1.238

Issue

There is a fundamental problem in preparing the repository which is not brought out in the statement. The inability to test non-destructively means that the boreholes drilled to characterize the repository area make it less suitable for a repository. (35)

Response

Tests at a candidate repository site should not proceed to such a degree that they would compromise the integrity of the site. Geophysical and borehole techniques are generally available to analyze geologic formations. Non-destructive sonic and a variety of other testing methods can also be utilized as reasonably effective tools to locate voids, fractures or even previously undetected drill holes. The need for additional research and development to improve and provide new methods for determining geologic and hydrologic properties is recognized in Section 5.2 of the final statement.

Draft p. 3.1.238

Issue

One commenter felt it should be pointed out here that satisfactory shaft and tunnel sealing techniques have not been developed. (218-DOI)

Response

Shaft and tunnel sealing is an important part of the total Repository Sealing Program. Most of the field work on the RSP has been done on boreholes but much of that work will have application in sealing shafts and tunnels. Work specific to shafts and tunnels will commence in FY-81.

Draft p. 3.1.238-239

Issue

There has already been considerable study of thermal effects which leads one to believe that subsidence can be calculated. (35)

GEOLOGIC CONSIDERATIONSResponse

A general discussion of the thermal and radiation induced effects on the host rock are provided in Section 5.2 of the final Statement. Calculations of subsidence or doming are dependent upon parameters such as repository design, host rock, waste type, waste form and package, etc.. Such parameters are of a site specific nature and would be more appropriately considered in a site specific evaluation.

Draft p. 3.1.242

Issue

The study of rock-waste interactions should include the geochemistry. Mobility of a number of radionuclides is strongly affected by the geochemistry (particularly the oxidation-reduction potential of the repository and ground water) and by the potential presence of complexing agents. These should be included in the proposed reasearch program. (113-EPA)

Response

This concern is presently under study for the Department by the Office of Nuclear Waste Isolation (ONWI) through the Waste Rock Interaction Technology (WRIT) Program (See Appendix L of final Statement).

Draft p. 3.1.243

Issue

It should be pointed out here that sorption phenomena (or "Kds") are not yet well understood and characterized. Considerable fundamental and field research is needed in these areas for both near- and far-field analyses and modeling. (218-DOI)

Response

During preparation of the final Statement this section was revised. Section 5.2 of the final Statement presents a discussion of the current R&D needs; among which the need to improve data gathering techniques for specfic rock properties and improved modeling methods are mentioned. R&D projects which are currently underway and are designed to meet these needs are identified in final Appendix L.2.

GEOLOGIC CONSIDERATIONSDOE/ET-0028, p. 7.2.2Issue

The statement that: "The repository should not be sited in or near an area in which igneous or volcanic activity has occurred during the post Pleistocene" should be assessed and actively discussed by DOE. An assessment should be made of the potential for volcanic activity and its impact on repository performance. The assessment should estimate the actual effects, detrimental or beneficial, on repository performance by different types of eruptions. (208-NRC)

Response

Acres of volcanic activity are highly unsuitable and our current abilities to predict the effects of such events are poor. Although distant volcanic activity and related seismic events are not expected to directly disturb a repository, these processes may significantly affect the regional hydrology which could provide a transport mechanism and pathway for radionuclide migration to the biosphere. Preliminary assessments evaluating the effects of volcanic activity on a repository have been performed. Although the results show beneficial effects in some cases, other cases show negative effects as mentioned above.

DOE/ET-0028, p. 7.2.3Issue

The credibility of Section 7.2 is weakened by either a lack of documentation for the Statements (e.g., see p. 7.2.6 Southwest Florida) or the use of very old references (e.g., see p. 7.2.6 paragraph 3 on the Supai Formation of the Holbrook basin of Arizona) when more recent material should be available. (208-NRC)

Response

The section in question utilizes as one of its references a USGS open file report that the commenter previously had requested to be cited because of it being an original reference.

DOE/ET-0028, p. 7.2.3Issue

The geologic term "formation" is misused throughout the GEIS. Although this appears to be a minor editorial comment, it may have legal ramifications. The term is defined in The American Code of Stratigraphic Nomenclature which is to be found in the Bulletin of the American Association of Petroleum Geologists (1961, pp. 645-660). Basically, a formation is a specific rock unit which has a distinctive lithologic characteristic which allows it to

GEOLOGIC CONSIDERATIONS

be mapped. Sandstone, limestone, shale, granite and basalt are not formations, whereas rock bodies such as the Dakota Sandstone, Salem Limestone, and Pierre Shale, and Louann Salt are. (208-NRC)

Response

The improper use of the term "formation" will be corrected upon publication of an addendum or an errata to the DOE/ET-0028 and DOE/ET-0029 documents.

DOE/ET-0028, p. 7.2.9

Issue

The statement that igneous rock "...range is chemical and mineralogical composition from granite to closely related rocks such as granodiorite" is technically true but misleading. The range goes far beyond granodiorite through gabbro to pyroxenite and dunite. (208-NRC)

Response

The statement identified by the commenter identifies the range of granitic rocks receiving detailed analysis as potential host rocks. The statement is thought to be appropriate and is not expected to be changed.

DOE/ET-0028, p. 7.2.9

Issue

The statement that "granite is mostly composed of silica and mica" is misleading. Mica makes up a small percent of most granites and quartz rarely exceeds 30%. Mention should be made of other minerals common in granite such as the feldspar and ferromangesian minerals. (208-NRC)

Response

The DOE agrees that the statement as it stands is misleading. Quartz and mica minerals are among the major minerals making up granite but they do not necessarily constitute the majority granite composition. The appropriate corrections will be made upon publication of an errata or addendum to the DOE/ET-0028 and DOE/ET-0029 documents.

DOE/ET-0028, p. 7.2.10

Issue

The basic references of Pirsson 1947 and Gilluy, Woodford and Aateus, 1968 should be replaced by reference to one of the following: Robert L. Folks Petrology of Sedimentary

GEOLOGIC CONSIDERATIONS

Rocks (Hemphill's, Austin, Texas), Blatt, Middleton, and Murray's Origin of Sedimentary Rocks Prentice-Hall or Pettijohn's Sedimentary Rocks, Harper Borthers, N.Y.. (208-NRC)

Response

In revising the draft Statement more current geologic references were used and appropriately cited. Where deemed appropriate, additional references would be added through publication of an errata or addendum to the DOE/ET-0028 and DOE/ET-0029 documents.

DOE/ET-0028, p. 7.2.10

Issue

Contrary to line 5, Table 7.2.1 gives no direct information on the mineral content of shales. (208-NRC)

Response

The appropriate changes will be made upon publication of an errata or addendum to DOE/ET0028 and DOE/ET-0029 documents.

DOE/ET-0028, p. 7.2.12

Issue

The statement that basalt is an "extrusive volcanic mafic (high in magnesium rock silicates) rock" is doubly misleading: (1) Not all basalts are extrusive e.g., Palisades Sill, and (2) mafic minerals are not limited to magnesium silicates. (208-NRC)

Response

Basalt is commonly described as an "extrusive volcanic mafic rock". However, in the interest of being technically correct, the description of basalt used will be revised upon publication of an errata or addendum to the DOE/ET-0028 and DOE/ET-0029 documents.

Issue

Because geologic historians suggest a rather constant transition and breakage of the earth's crust, the probability of a major disruption in the storage of at least some of the radioactive material is quite high. (73)

Response

Crustal breakage generally occurs in belts of activity along crustal plate boundaries or in tectonically active areas. The site selection process will avoid all areas of known

GEOLOGIC CONSIDERATIONS

tectonic activity and look for areas of stability, such as some of the salt basins which contain salt that has been in existence and essentially undisturbed for several million to hundreds of millions years.

Issue

The effectiveness of the host rock to act as a barrier is dependent upon site-specific parameters and cannot be attested to at this time. (43, 97)

Response

A potential host rock has certain physical properties that make it favorable for a repository host rock. These properties are inherent in the rock and are typical of the rock. The geologic setting may contain elements related to structure, seismicity, physiography, etc. that modify the effectiveness of the site at a given location. The site selection process will seek out sites that are not modified by these elements and where the host rock properties will be intact.

Issue

If the repository were located in a salt bed, the ground-water would first have to dissolve tremendous quantities of salt prior to picking up and transporting the small amount of radioactive material remaining after 600 years. Would not the salt do far more damage to the environment and the local hydrology in particular than the small quantity of transuranics within the facility? Is not worrying about reprocessed nuclear waste beyond 600 years in a salt bed akin to worrying about arsenic being spiked with traces of a poison? Please discuss this concept in your draft. (178)

Response

Should ground-water contact a salt bed containing a nuclear waste repository, long times and large quantities of water would be required to uncover the waste. The ground water leaving the repository would be saturated with salt and would have to be diluted about 300 to 1 in order to be potable. The damage to the environment from both the salt and the radionuclides is expected to be small. Which is smaller depends on the conditions of the scenario and the definition of "damage".

Issue

One commenter mentioned that the Sierra Club Bulletin asserts: "The temperature within the (salt) repository may reach 300°C. Water, in the form of liquid and vapor, is drawn towards the heat source in a salt repository...This hot brine solution is acidic and very corrosive. According to EPA, the canisters would be breached in a decade or less" (Sierra July/August 1979, p. 51). The data relied on in the draft Statement are taken at room temperature. (153)

GEOLOGIC CONSIDERATIONSResponse

Brine migration of liquid already present in the salt (0.1 to 1% by weight) and its migration up the thermal gradient toward the waste package is under extensive study. If the environment produced by elevated temperatures and the presence of brine have an adverse effect on the performance of the waste package, then materials must be selected which can withstand this environment. Data from package material design and performance testing would be used for package design and material selection to insure optimum waste package performance.

Issue

There are many problems specific to salt as a repository medium that should be addressed. (42, 68)

Response

Problems specific to salt as a repository material (e.g., thermal effects, lack of sorption, closure rates, corrosive brines) are addressed in generally qualitative terms in Section 5.2. There are additional data in other reports (ERDA 1976b and Brandshaw et. al. 1971). These have been incorporated into Section 5.2. The DOE Position Paper to the NRC rulemaking proceedings on nuclear waste storage and disposal (DOE 1980a) also contains an extensive discussion of the environmental conditions in a salt repository.

Issue

One commenter noted that linear thermomechanical analysis is applied to a repository in salt. Such an analysis can result in significant error in predicting thermomechanical effects. Even with this analysis a surface uplift up to 1.5 m is predicted. The important question not addressed in the GEIS is what effect will this have on shaft and borehole seals, thermally driven convection and breccia pipe formation? (208-NRC)

Response

The NWS program has developed analytical and numerical solutions, and material properties to apply non-linear thermomechanical analysis to the behavior of a repository in salt. These solutions and properties have been used to model the behavior of field experiments such as Project Salt Vault and Avery Island. These studies are continuing and the accuracy of modeling or predicting the behavior is expected to be improved. These improved models will be used to analyze the total behavior of a potential repository in salt.

Issue

"(Creep) is difficult to stabilize in tunnel openings."

"From experiments...equations can be developed to describe the creep behavior of salt."

GEOLOGIC CONSIDERATIONS

Since equations have been developed which describe the behavior of salt a posteriori, the GEIS should discuss whether they can predict the behavior of salt under thermomechanical loading conditions. (208-NRC)

Response

Some amounts of creep can be tolerated with no adverse effects in a repository. Excessive creep is not desirable and does present special engineering challenges. Stabilization of openings against creep is generally possible.

The NWTs program includes study of existing data and ongoing laboratory and field tests to obtain the creep and other thermomechanical properties of salt. With this data and the analytical studies also underway, it is anticipated that relationships will be developed to adequately describe the creep behavior of salt. It has already been demonstrated, for example, that the result of the Project Salt Vault field tests can be predicted to a reasonable accuracy with the materials properties and analytical solutions now available. Predictions of the behavior of salt under anticipated thermomechanical conditions will be included in repository design studies.

Issue

It was noted that in view of recent news articles from Mississippi reporting accidental releases of radioactive material from weapons testing sites, how does DOE view the integrity of salt as waste repository media? (43)

Other commenters pointed out that disposal in salt repository will not work because 1) salt is often found near mineral deposits, 2) of drilling concerns, 3) presence of brines will cause dissolution of waste form, 4) salt has low sorptive capacity, and 5) elevated temperatures and pressures and the presence of water will compromise waste form. (62; 197)

Response

Some contamination by tritium has been observed at the Tatum Dome in Mississippi where two test explosions of nuclear weapons took place in the mid-1960's. This contamination has however been shown to be related to disposal of tritiated water into surrounding aquifers as part of the clean-up operations. Geologic repositories, as described in this Statement, would only receive radioactive wastes in solid form. Contamination levels reported at the Tatum Dome were caused by specific operations and practices completely different than those to be undertaken in geologic repositories.

Salt is considered as a disposal medium because it has potentially useful properties (plasticity, isolation from flowing water, etc.) and because of its stability. In the Statement it is treated as one of the four geologic materials that are being considered as candidate repository media. Salt has had relatively more investigation than the other three media types at this time, but as with the others, it requires continued study and consideration.

GEOLOGIC CONSIDERATIONS

The preliminary site selection process (Section 5.1) considers such factors as resource potential, proximity to natural resources, tectonic stability, water system, etc. The Statement considers that repositories are potentially feasible in all media. However, the Statement acknowledges that there are some unresolved issues regarding these media and suggests continued research and development to fill existing data gaps.

Issue

The statement that granite has "...little ability to deform under stress" is not true. Under varying combinations of the following: 1) high confining pressure, 2) elevated temperatures, or 3) when the stresses are applied for long time spans, granite will deform. (208-NRC)

Response

Granite will undergo deformation under the influence of pressure, time, and temperature. For the levels of these phenomena predicted in a repository, however, such deformations are expected to be small and certainly manageable.

The deformation characteristics of granites are being studied analytically, in the laboratory, and in field experiments approximating repository conditions. Information obtained from these studies will be used to evaluate deformations in, and prepare designs for, a repository in granite.

Issue

The statement that the "mineral components of granite are almost inactive chemically under ambient temperature and pressure conditions" is misleading. Granite does decompose at surface temperatures and pressure as evidenced by well developed regoliths found on top of many granites. (208-NRC)

Response

Given the anticipated time frame (on the order of several tens of years) for exposure of granitic rocks to ambient temperature and pressure in a repository excavation, it is reasonable to assume that the mineral components of granite would be relatively inactive chemically. Decomposition of granite into well developed regoliths under conditions of surface temperature and pressure is a phenomena requiring periods of time orders of magnitude greater than those expected for similar physical conditions in a repository. In addition, regolith formation is assisted by constant exposure to the hydrologic cycle. Repository design will be such that moisture in the excavation will be kept to a minimum.

GEOLOGIC CONSIDERATIONSIssue

Several commenters stated that the final Statement should address the interrelationship between deep and shallow ground-water aquifers and surface water systems and the potential for transport of nuclides between these systems. (43, 97)

Response

The relationship between subsurface aquifers and surface waters will become an issue when a proposed site is under investigation for site qualifications. The ground-water/surface water conditions and the associated physical parameters are likely to be different for each site and specific to each site. The potential for transport between the surface and ground water systems does exist, but the investigations and site qualification would look specifically at questions of this type. The particular set of physical conditions that will determine how effectively the water systems are isolated from each other will need to be investigated for each site and an assessment of transport potential made at that time. See Section 5.1 for a related discussion.

Issue

The Statement should address the unique geophysical characteristics of nuclear-overstressed caverns. (115)

Response

The Statement is directed specifically toward geologic disposal in openings formed by conventional mining and/or drilling methods. Nuclear-overstressed deep caverns are not considered for this reason.

Issue

"The heat from nuclear wastes will induce stress in the hot rock" and the significance of these stresses is uncertain as is the temperature effect on the rock properties. (98)

Response

The study of the thermomechanical behavior of rock is included in the NWTS program. This program includes laboratory experiments, field observations, and analytical and numerical solutions.

Laboratory experiments are proceeding at such institutions as University of Minnesota, Texas A&M, and University of California. Field testing of the thermomechanical behavior of rock is being conducted in Sweden (Stripa granite); at the Nevada Test Site (Climax quartz monzonite); in Kansas (Project Salt Vault); at Avery Island in Louisiana; and in Washington (basalt). It is expected that this integrated program will provide the necessary understanding of heat-induced stresses in rock, and of the thermomechanical response of the rock.

GEOLOGIC CONSIDERATIONS

Issue

Radiolysis will cause hydrogen and oxygen to form in a bedrock cavern, thus creating a potentially explosive atmosphere. Should an explosion occur inside the cavern, the consequences are really unknown. It will place stress on the cavern and the aquifer and increase the chances of water movement, thus increasing the potential for additional contamination of the aquifer. The authors fail to address such possibilities. (97)

Response

The consequences of an explosion were not addressed because the probability of one occurring after emplacement of waste is considered to be infinitesimally small. The explosive range for hydrogen in air is 4.1 to 74.2 volume percent. Mixtures in this range will ignite and burn if the temperature is above the ignition point, 1085°F (585°C) (Perry 1950). Temperatures will never be this high. Below this temperature, a spark could cause an explosion in mixtures in the explosive range. This will never be achieved in an open storage room (e.g., in salt; 5.5 m x 6.7 m x 1070 = 39,400 m³), because a production of 1615 m³ (0.041 x 39,400) of H₂ would be required. Production of this much hydrogen would require the complete radiolysis of 1300 liters of water with no recombination or the total corrosion of 3,000 kg of iron by an aqueous corroding medium.

Preliminary estimates of the quantities expected by radiolysis of brine (or water) in the backfill surrounding waste packages were performed by Jenks (Jenks 1980). His work indicates that such accumulations will not be approached by even an order of magnitude. Jenks (Jenks 1979) also made estimates of corrosion accelerated by radiolysis which also indicates that adding hydrogen produced by aqueous corrosion will still not cause the quantity to approach that needed for the explosive range.

The explosive range could possibly occur in voidage in the backfill surrounding the waste package. Sparks are not likely in this location. In any case, the reaction could not be propagated through the backfill because the baffling effect of the backfill particles.

Issue

Additional geologic background information should be provided in the Statement, and the discussion of tectonic effects should be expanded. (10)

Response

This Statement is generic in nature, and the geologic history of formations (salt, basalt, granite, shale) and plate tectonics are discussed in general terms. Detailed discussions would be more appropriate and will be included in future site-specific EISs.

GEOLOGIC CONSIDERATIONSIssue

It will take at least fifty years to really assess the effects on a geologic medium, further expansion of a site already in operation would be just asking for trouble. (164, 187)

Response

Design of a repository will include the best available information from laboratory studies, field observations, and analytical and numerical solutions. After construction of a repository begins, it will be monitored to obtain information on its behavior with time.

Issue

The plugging and sealing of shafts, tunnels, and boreholes should be discussed more extensively. (219)

Response

The subjects of plugging and sealing shafts, tunnels, and boreholes are discussed in detail in an Office of Nuclear Waste Isolation (ONWI) document (ONWI 1979).

Issue

It was pointed out that programs on the radiation effects of other minerals (than rock salt) should be initiated and/or expanded. (25)

Response

The effects of irradiation on inorganic materials, including many minerals, have been known for many years and are generally small, particularly with respect to mechanical properties. Although the effects on candidate host rocks are expected to be of only secondary concern, a program to investigate such effects has been underway for several years. Initial emphasis was on rock salt but other materials are included, e.g., granite and basalt. As considerable data has been accumulated on rock salt, the program emphasis is now shifting to the latter materials. Also, work involving candidate waste forms, canister materials, etc., and a variety of ground waters in the presence of radiation has been underway for some time. The effects of radiation are considered in screening candidate materials. Since interactions among materials in the presence of radiation is strongly dependent on waste package design, choice of materials, and the nature of the host rock, these tests will increase as site specific package designs become better defined.

Documents addressing radioactive waste-induced effects include:

- ERDA 76-43, Alternatives for Managing Wastes from Reactors and Post Fission Operations in the LWR Fuel Cycle, 1976.

GEOLOGIC CONSIDERATIONS

- Brandshaw, R. L., and McClain, W. C., Editors, Project Salt Vault: A Demonstration of the Disposal of High Activity Solidified Wastes in Underground Salt Mines, ORNL-4555, Oak Ridge National Laboratory, Oak Ridge, TN, April 1971.
- DOE/NE-0007, DOE Position Paper to the NRC Rulemaking on Nuclear Waste Storage and Disposal, April 1980.

Issue

It was suggested that the effect of radiation on the four repository media (salt, basalt, granite, shale) should be discussed. (25)

Response

Data on the subject of radiation effects on media is very limited. There is some information for salt, but almost nothing for granite, basalt, and shale. The effects of radiation on potential geologic host media have generally been considered to be of secondary importance (DOE 1980a). Information from comment letter #25 (Levy, Brookhaven National Laboratory) has been integrated into Section 5.2.

Issue

One commenter noted that the Statement should contain more discussion of (or information on) the geology and exploration effort required to support conventionally mined geologic disposal. (12)

Response

From a generic standpoint for the Statement, candidate regions for the four rock type disposal media were selected based on rock properties, known occurrence, and suitability for conventional mining techniques. This was done primarily from available literature and no further breakdown to smaller areas was made. The description of a site selection and qualification process is given for the time if and/or when a medium is selected and further investigations are begun. From this time on geologic exploration and data gathering become increasingly important at each progressive level until a site is accepted or rejected.

The DOE Position Paper to the NRC rulemaking proceedings on nuclear waste storage and disposal (DOE 1980a) presents the status of the on going geologic exploration programs and discusses in depth the site exploration, characterization, and selection process.

Issue

The underground firing of nuclear explosives results in the formation of vitrified debris, because of the solidification of molten and vaporized rock. Thousands of tons of such vitrified debris have been in place for periods of up to 25 years, mostly in tuff at

GEOLOGIC CONSIDERATIONS

the Nevada Test Site, but also in granite, shale (Gas Buggy), and salt. This experience bears directly upon the proposed long-term storage of vitrified high level waste, and should be discussed. (208-NRC)

Response

A discussion regarding the investigation of underground nuclear tests relative to the migration of radionuclides can be found in a separate report (Ramspott 1978).

Y/OWI/TM 36/21Issue

This document addresses only three host rock media - granite, basalt and shale. No basis for the apparent conclusion that groundwater movement in salt is negligible has been presented in either GEIS or in TM-36. Note also that the permeabilities of granite and basalt presented in the GEIS (Table 3.1.1, p. 3.1.9) are nil and therefore the repositories in granite and basalt could presumably be located at depths significantly less than salt and shale. (208-NRC)

Response

Appendix B of the final Statement (and draft Section 3.1.1) point out that existence of salt formations that are estimated to be hundreds of millions of years old testifies to their stability and their isolation from water. The statement regarding permeabilities of granite and basalt being nil has been modified (see final Appendix B).

MULTIBARRIERS FOR DISPOSALDraft p. 1.5Issue

One commenter suggested that the discussion of multiple barriers should include the barrier-like effect of liquid transport that result in dilution and dispersal, even though these processes are technically not barriers. (208-NRC)

Response

Dilution and dispersal imply fluid flow into or through the repository. The multiple barriers are meant to reduce or prevent fluid flow and/or transport of waste. It would seem paradoxical to assume benefits from the processes the barriers are intended to prevent.

Draft p. 1.15Issue

Since some bentonites lose water above 100°C, perhaps illite which does not lose water above 100°C should be added as a potential overpack material for canisters. (113-EPA)

Response

The statement, "Absorptive overpack materials such as zeolites and bentonites..." is illustrative of types of materials considered and available for canister overpacks. The examples given are not recommended as the final or only ones.

The statement was modified to read "... bentonites/illite..."

Draft p. 3.1.1Issue

Multiple barriers are listed as one of six characteristic for geologic disposal when, in fact, three of the other five characteristics are themselves important barriers to nucleide transport. (218-DOI)

Response

The six items listed are characteristics of a mined geologic disposal system. Item 6 was identified in order to emphasize the desirability of the combined (redundant) effects of the individual barriers.

MULTIBARRIERS FOR DISPOSALDraft p. 3.1.6Issue

One commenter stated that multiple barriers should not "act together" but independently, so that if some fail the others compensate. (218-DOI)

Response

Multiple barriers are intended to act independently to prevent waste migration and enhance isolation. The wording in the final Statement was changed to reflect this (see Section B.6 of the final Statement.).

Draft p. 3.1.15Issue

Concerning the proposed Swedish Containers, no appraisal is made as to whether this is felt to be a good design and an advancement on the state of the art. If it is, it should be so stated. If not, it is not clear why it is mentioned. (34)

Response

Section 5.1.2.3 of the final Statement presents a discussion of the Swedish approach to waste package design. This section notes that the Swedish work did a great deal to promote acceptance of the multibarrier waste package and that although the Swedish design may be somewhat more complex than others presently being studied, the Swedish designs have increased understanding of long term package performance.

Draft p. 1.15Issue

The proposed Swedish canister is not "highly" sophisticated; it is a simple copper can with lead fill. The engineered sorption barriers are not part of the canister but part of the backfill buffer around the canisters. In the last reference it is not clear what "redox materials" are. (218-DOI)

Response

As noted above, final Section 5.1.2.3 presents a revised discussion of the Swedish approach to waste package design. The final Statement uses the terms waste package or waste package system when discussing the waste canister and the overpack material (See Section 5.1.2). Redox materials are materials that could influence the oxidation-reduction potential of the repository system.

MULTIBARRIERS FOR DISPOSALDraft p. 3.1.17Issue

Several uncertainties exist in the projected behavior of the system, such as the philosophy for radionuclide containment, waste form, the host rock, etc. Suggest a discussion as to how these uncertainties can be overcome. (113-EPA)

Response

The multiple barrier or multi-barrier discussion has been expanded (see Appendix 5.1.2) from what is in the draft and will add justification for the "defense-in-depth" philosophy quoted on p. 3.1.17 of the draft. The sections "Status of Technical Development and R&D Needs" discuss the uncertainties for each disposal concept and current efforts being made to resolve them.

Draft p. 3.1.39Issue

One commenter stated that item 3 under Performance Criteria appears to be a listing of conditions requiring consideration rather than criteria. (58)

Response

The intent was to list conditions requiring criteria to be developed; they in themselves are not criteria.

Draft p. 3.1.40Issue

Does the waste package design refer to the container alone, the container plus waste or the entire system? (113-EPA)

Response

For mined geologic disposal (including drilling) the waste package is described as the waste (form and material), any material between the waste and canister wall, the canister, and any overpack material around the canister or between the canister and the host rock.

MULTIBARRIERS FOR DISPOSALDraft p. 3.1.40Issue

It is recommended that the bulleted items "Licensing" and "Cost/Benefit Issue" be removed, since those issues do not pertain to the technical feasibility of waste packaging. (124)

Response

DOE agrees. This has been done.

Draft p. 3.1.48Issue

Anisotropies in the rock body are identified (bedding, etc.). This is contradictory with the avowed goal of a homogeneous host rock. Anisotropies, whether in horizontal or inclined units are anisotropies. Even in horizontal units, lateral anisotropies are common. Horizontal bodies may have greater roof problems than an equivalent weakness along the foot-wall of the repository. (43)

Response

It is true that isotropy in some property(ies) is common in rock units, particularly in bedded or laminated units, but this is not contradictory to homogeneity of the same unit. Many sedimentary rocks are homogeneous and isotropic with respect to some property a unit is homogeneous if the property(ies), isotropic or anisotropic conditions are constant over the unit. (See Davis and DeWiest 1967.)

Draft p. 3.1.54Issue

Waste packaging can be important for the first 1000 years. Paradoxically, after that it would be better to allow the waste to move through the nearby host rock, thus reducing the concentration and increasing the isolation. (154)

Response

Some migration analyses have been made on which it was assumed all of the waste dissolved at 1000 years. Isolation is not increased. Depending on the assumptions made concerning flow rate of water, path length to the biosphere and the types of geological formations that the water passes through different amounts of radioactivity ultimately reach the biosphere.

MULTIBARRIERS FOR DISPOSALDraft p. 3.1.54-62Issue

One commenter stated that since glass is already an adequate waste form, there does not seem to be any need to emphasize the need for "additional experimentation." The commenter also stated that it is unfortunate that calcine, which has more real operating experience behind it than any other waste form, is given scant praise. (154)

Response

It has not yet been established that glass is a fully acceptable waste form. For example, proposed NRC performance criteria, 10 CFR 60, E, (published 5/13/80) would require reasonable assurance that radionuclides be contained for at least 1000 years. Research is required to provide this assurance. Other experimentation is that normally associated with taking a process from a pilot plant status and scaling it up to a full-scale operating plant.

Concerning calcine, the operating experience has been with wastes that are generally over a factor of 100 less radioactive than commercial HLW and that have a quite different chemical composition. Granular calcine may be a satisfactory defense waste form for the long-term but it would probably have to be consolidated by incorporation in a matrix or by pressing and sintering before it would be a suitable HLW form. These latter processes have been studied to a limited extent and are discussed in Section 4.3 of the final Statement.

Draft p. 3.1.55Issue

The leach rates of spent fuel in typical groundwaters at temperatures to be expected in spent fuel repositories are more important than leach rates of spent fuel in room temperature deionized water. (113-EPA)

Response

The comment is correct and such leach rates are being obtained in the ONWI-sponsored Waste Rock Interactions Technology program at Battelle-Northwest Laboratories. However, leach rates obtained in room temperature deionized water are not irrelevant. They furnish a good initial reference point for comparison, and experience has shown that they do not differ from groundwater by more than a factor of ten at the same temperature.

MULTIBARRIERS FOR DISPOSALDraft p. 3.1.55Issue

The correlation between leaching of Zircaloy-clad fuel elements and referenced leach rates of unclad UO_2 pellets is very misleading. Explain why no credit is given to containers. (181)

Response

When full safety analyses are made, credit is usually given to the container by assuming that a certain time period elapses before the onset of leaching. The time period varies depending upon the scenario being studied. But, to be conservative, when leaching begins it is assumed that the cladding has "disappeared", i.e., no credit is taken for the protection provided by residual intact portions of cladding.

Draft p. 3.1.56Issue

There is a discussion of the behavior of glass in hydrothermal environments which appears to ignore the potential for simply eliminating the problem by reduced waste loadings rather than extensive research studies. (198)

Response

Repository and waste package designs will be technically conservative to the extent that hydrothermal conditions will not occur under normal, anticipated repository conditions. Studies are underway to define abnormal scenarios that could lead to hydrothermal conditions if indeed they are possible. If so, the potential risk must be assessed and evaluated relative to trade-offs such as reduced waste loadings, improved waste form development, further aging of the waste prior to disposal, etc. The resultant decision must consider the impact on the total waste management system.

Draft, p. 3.1.56Issue

Has the Eh-pH dependency of the waste form been investigated? The waste itself, having multiple oxidation states, will have different solubilities with differing Eh-pH. Can we adequately characterize the Eh-pH of groundwater after they have reacted to some extent with well rocks? We are not talking of a hypothetical distilled water interaction. Appendix I does not seem to consider water quality. (43)

MULTIBARRIERS FOR DISPOSALResponse

The properties of the waste form, both physical and chemical, are being carefully considered so that the waste form selected will be best suited for the respective disposal environment. The Eh-pH of groundwater can be measured with sufficient accuracy to permit modeling the interaction between groundwater and waste form following disposal in a geologic repository.

Due to the site-specific nature of groundwater chemistry, it was not felt appropriate to have a detailed discussion of groundwater chemistry in this Statement. Such discussion and analysis will be provided in any future site-specific analysis.

Draft p. 3.1.59Issue

It seems impossible to maintain canister integrity for a significant time period because disruption during gas, oil or mineral exploration may destroy the canister.
(113-EPA)

Response

This comment discounts probability. There is a low probability that exploration will even occur since the repositories will be located in areas selected as having negligible commercial mineral value. (Salt repositories are an exception but salt is so widespread and abundant there remains a low probability a given site will be selected for commercial exploitation.) Even if drilling intrudes a repository, there is a low probability that a canister will be affected because only a small amount (<0.001) of the cross-sectional area is occupied by canisters.

Draft p. 3.1.59-62Issue

Several commenters suggested various packing materials (zeolite, bentonite) be considered. (17, 154) Others questioned the effectiveness (sorptive qualities) of overpack at elevated temperatures (43,97).

Response

The final Statement contains a discussion of many different kinds of engineered barriers that are being considered (see Section 5.1.2).

It is agreed that additional R&D must be performed on absorptive barrier materials. Retardation of radionuclides should be a goal but the materials selected should be optimum for all possible events and scenarios which initiate radionuclide migration.

MULTIBARRIERS FOR DISPOSALDraft p. 3.1.61Issue

One commenter stated that the stainless steel/lead/titanium composite canister is undoubtedly very expensive. Titanium is not exactly plentiful either. Mass production of these canisters may become prohibitive. (35)

Response

Using the canister description on p. 3.1.61 of the draft Statement, the cost of a steel/lead/titanium canister was estimated as follows:

lead requirement per canister	1,800 kg @ \$1.10/kg	\$ 2,000
fabricated titanium shell	108 kg @ \$28/kg	\$ 3,000
fabricated 30 cm dia. stainless steel canister		<u>\$ 5,000</u>
total material cost		\$10,000
additional fabrication and other costs		\$ 5,000
TOTAL		\$15,000 or \$ 5/kg HM

The total cost of \$5/kg HM for such packaging agrees well with EPA estimates of 4-6 \$/kg HM. (EPA 1977).

The incremental cost over that stated in the document is about \$3/kg HM in a total waste management cost of approximately \$160/kg HM or about two percent. Thus, use of such canisters does not appear to be prohibitive from an economic standpoint.

In a 250 GWe system, approximately 80,000 canisters of high level waste would be generated. This would require a total of 8,600 MT of titanium metal or less than half of one years average annual production for the period 1968-72 (DOI 1972). Availability, therefore, also does not appear to be a limiting factor.

Draft p. 3.1.62Issue

The statement is made that, "There apparently has been no information published on the corrosion resistance of the glass ceramic material under repository -- simulating test conditions." However, it is also stated (Table, p. 3.1.60) that glass ceramic canisters have an estimated life beyond 5,000 years. (9, 36)

Response

The statement is in reference to the Swedish program. The table was removed.

MULTIBARRIERS FOR DISPOSALDraft p. 3.1.62Issue

Discussion of the Swedish waste disposal study does not acknowledge the negative comments made by the California State Energy Commission, the U.S Geological Survey and the Jet Propulsion Laboratory in their review of the KBS Safety Analysis. (55).

Response

Comments were sought from experts in many countries before the Swedish government accepted the feasibility of the KBS waste management concept. It was recognized that there will always be uncertainties in predicting long-range future behavior of materials and geological formations, but it was the concensus of the experts that the KBS concept reduced the uncertainties to such a low level that the viability of the proposed waste disposal system was assured with satisfactory confidence.

Draft p. 3.1.62Issue

The likelihood that oxygen will be introduced into the repository when it is constructed and therefore be available to the groundwater should be considered in evaluating canisters and the mobility of some nuclides. (113-EPA)

Response

The technical data used in evaluating canister corrosion and nuclide mobility has almost all been obtained with oxygen (air) present, thus the introduction of oxygen in the repository is being considered. Experimentally, the bigger problem is to eliminate oxygen in order to study the conditions expected after a repository has been sealed for hundreds of years.

Draft p. J.6Issue

Figure J.1 should be explained. Its applicability is unclear. (113-EPA)

Response

This figure does not appear in the final Statement.

MULTIBARRIERS FOR DISPOSALDraft Appendix LIssue

In Appendix L (the draft statement) the statement that devitrified glass is stronger than ordinary glass and will resist further fracturing is not as important as the potential greater leaching from devitrified glass. (113-EPA)

Response

DOE believes the relative importance of these two factors depends on the circumstances being considered. For instance, the importance of fracturing behavior probably predominates in a transportation accident while leaching behavior will be more important after emplacement in the repository. The essential information that appeared in Appendix L of the draft Statement has been incorporated into Chapter 4 of the final Statement. Additional details can be found in DOE/ET-0028.

Draft Appendix PIssue

Ringwood and co-workers have identified a suite of minerals for use in waste disposal. Their work should be referenced and seriously considered. (113-EPA)

Response

Ringwood's work is referred to in final Section 4.3.2.

Draft Appendix OIssue

This is a rather interesting appendix although the development of the field does not appear to be sufficiently advanced for any convincing environmental impact analysis. There seems to be a contradiction between Tables 0.4 and 0.5. In Table 0.4 a 1 meter barrier is reported to retain strontium-90 and cesium-137 for about 30 years, or about one-half life for these nuclides. In Table 0.5 a 1 meter barrier is said to retain them for 30 half lives.

The possible competition for ion exchange sites on added minerals (or natural minerals for that matter) should be noted. Canister materials are elements of the transition series, notably iron, nickel, chromium, or titanium. In Sweden, lead and copper have been suggested for canisters. The ion exchange capacity of any added materials must be enough to handle the nonradioelements as well as the radioelements. (113-EPA)

MULTIBARRIERS FOR DISPOSALResponse

This appendix has been deleted from the final Statement.

Issue

Several commenters stated that a more thorough treatment of the possible contribution of natural and engineered barriers should be provided in the final Statement together with a summary description of the related R&D program(s). (43, 58, 97, 114, 124, 154, 219)

Response

DOE agrees that the concept of multiple barriers and their use in future repository systems was incompletely described in the draft. Further discussion of multiple barriers and the associated R&D efforts may be found in Sections 5.1 and 5.2 of Volume 1, respectively.

Issue

Several commenters noted that the Statement should make clear whether or not glass is an acceptable high-level waste form for the initial repository. (58, 124)

Response

The acceptability of a particular waste form will depend upon the specifics of the final system design and repository medium. A discussion of the properties of glass as well as other possible waste forms is contained in final Section 4.3.

Issue

Several commenters questioned the effectiveness of engineered barriers and/or whether these barriers would be of significance over the long term. (6, 43, 113-EPA, 141, 142, 214)

Response

A systems (or multibarrier) approach to the design of waste repository assumes that the fate of radionuclides will be determined by geologic environment, the properties of the host medium, the waste form chosen, and any other engineered barriers utilized. In selecting these design components, the objective will be to prevent release of radionuclides to the biosphere by providing redundant as well as independent barriers. DOE recognizes that the engineered components of the multibarrier system would be of greatest importance in the short term and that the repository medium and the surrounding geology would be the critical elements over long periods of time. It is envisioned that the waste container will be designed to remain intact for about 1,000 years. The recent DOE Position Paper to the NRC Rulemaking proceedings on nuclear waste storage and disposal (DOE 1980a) notes that "Waste

MULTIBARRIERS FOR DISPOSAL

containment within the immediate vicinity of initial placement should be virtually complete during the period when radiation and thermal output are dominated by fission product decay."

Issue

One commenter noted that a single, standard canister design capable of being used in all media may be advantageous in regards to health, safety and cost. (97)

Response

DOE (and in turn the general public) might enjoy some cost advantages if a single standard canister design were to be used in all repository media. However, DOE believes that the development of the system components (i.e., canister, overpack) will be dictated by site-specific parameters and, therefore, these components will probably differ across repository media and from site to site.

Issue

Data are available on existing and proposed low release solid forms. These data should be presented and their significance analyzed. (219)

Response

Section 4.3.2 of the final Statement discusses a variety of potential waste form alternatives.

SOCIOECONOMIC/SOCIOPOLITICAL ISSUESDraft pp. iv and 1.13Issues

It is all right to spell out the technical requirements to be met in siting of repositories, but should it not be mentioned that the major impasse in site selection will be the gaining of acceptance of the population surrounding the proposed site(s)?

Sociopolitical factors should be approached in Stage I in the selection of regions. It seems more logical to gain regional acceptance before tying into the local acceptance issue. (181)

Response

Appendix B of the final Statement notes that the nontechnical concerns will be evaluated throughout the entire site-selection process. Also see final Section 2.3.

Draft pp. 1.6 and 1.15Issue

One letter suggested that institutional controls would increase isolation of waste but not containment. (154)

Response

It is believed that the effect of such control is apparent, however, as noted below, minimal reliance will be placed on long-term institutions.

Draft pp. 1.6, 1.15, 3.1.62-64Issue

One commenter stated that institutions of man would provide adequate repository markers to prevent inadvertant human intrusion. (17) Other commenters felt that waste management programs could not and/or should not rely on long-term transfer of information, and such mechanisms would not prevent human actions over the long term. (41, 43, 62, 113-EPA, 171, 186, 213, 216, 218-DOI)

Response

While this Statement concluded that there are apparently no reasons in principle why human surveillance could not survive for hundreds of years, it is also concluded that waste management systems adopted in the present should place minimal reliance on any human institutional management after repository closure (see final Section 3.5).

SOCIOECONOMIC/SOCIOPOLITICAL ISSUESDraft p. 1.6Issue

One commenter suggested that the discussion of the relative permanence of human institutions was not illuminating and could best be removed. (40)

Response

DOE disagrees and would note that the purpose of the discussion of human institutions (as well as other non-technical issues relating to waste management) admittedly was to air such issues. Such issues will, however, have to be resolved in any on-going program.

Draft pp. 1.6, 21, 22Issue

Several commenters questioned whether repository markers would last for millenia in view of the possibility of vandalism, acts of war, natural processes. (55, 154, 208-NRC, 218-DOI)

Response

Such considerations would have to be included when designing markers. For non-salt repositories, rock remaining at the surface could be a starting point for a monument.

Draft p. 1.13Issue

Several commenters requested that the criteria be identified which would be used for site screening based on socioeconomic and sociopolitical factors. (43, 208-NRC)

Response

DOE has adopted a set of siting criteria (ONWI 1980) concerned with human population density as well as socioeconomic and political factors.

Draft p. 1.15Issue

One commenter stated that it is unacceptable to refer to historical examples which suggest that society can maintain certain systems over centuries and not to give examples. (40)

SOCIOECONOMIC/SOCIOPOLITICAL ISSUESResponse

Appropriate references have been included in the final Statement (see Section 3.5)

Draft pp. 1.15, 3.1.62-64

Issue

The section entitled Human Institutions calls attention to the merit of setting up such institutions in the long-term control of nuclear wastes. It would be helpful to address whose responsibility it would be to establish and maintain such institutions. (208-NRC)

Response

Institutional arrangement for regulation and control of the nuclear waste are the responsibility of the Federal legislature. At present the EPA and NRC have standards setting and regulating jurisdiction while the DOE has ownership control.

Draft p. 3.1.25

Issue

One commenter felt that the discussion of land use and transportation considerations does not adequately assess transportation demands. (43)

Response

The discussion on p. 3.1.25 of the draft Statement was intended to be only a general discussion of site selection criteria for a geologic repository. For a more complete discussion of transportation demands, see Section 4.5 of this final Statement.

Draft p. 3.1.25

Issue

Land use and transportation considerations are of concern only in the short term. (113-EPA)

Response

Compared to the lifetime of the repository, this material does apply to the near-term time period. However, these factors (land use and transportation) do need to be considered in the site selection process. They would enter into the acceptance or rejection of a site and are included for this reason.

SOCIOECONOMIC/SOCIOPOLITICAL ISSUESDraft p. 3.1.25Issue

The discussion in the section entitled Land Use and Transportation Considerations focused on some possible land use conflicts and refers the reader to a body of literature, some of which is described as speculative. It would be useful for the GEIS to summarize this information and to present it for review. (208-NRC)

Response

The question as asked is not understood. This particular section did not refer to a body of literature.

Draft p. 3.1.43Issue

In the case of enforcement against private organizations, criminal penalties could be imposed. (208-NRC)

Response

Regulatory agencies usually do not have power to impose criminal penalties. Violations of the criminal code, however, can be handled through usual criminal justice procedures.

Draft p. 3.1.43Issue

The following is a suggested additional paragraph under "Regulatory Tasks":

Monitoring. In view of the questions raised in technical and performance areas of geologic repositories, appropriate instrumental monitoring should be undertaken for a reviewing agency. The data should be directly received by the agency, copies should be distributed for analysis to technical groups within the agency or to contractors, and the agency should store the data for retrieval as required. Continuous, as well as periodic, analysis programs should be established by the agency. (154)

Response

Such a detailed monitoring plan is considered to be covered by the general functions of standard-setting and licensing (setting monitoring requirements) as well as inspection (receiving reports). Any specific approach such as this is beyond the scope of this document.

SOCIOECONOMIC/SOCIOPOLITICAL ISSUESDraft p. 3.1.44Issue

One commenter noted that in the second paragraph, the lack of direct benefits (from waste material) to future generations is discussed but the possibility of indirect benefits is relegated to a footnote. A balanced consideration of long-term risks and benefits must consider both. (58)

Response

This correction has been made in preparing the final Statement (see Section 3.5)

Draft p. 3.1.64Issue

The first century after closure of the repository would be critical for "hands on" corrective action only if the monitoring program established some deficiency in the repository. Although the radioactivity of the repository has been reduced substantially after 700 years, the threat is by no means negligible. (113-EPA)

Response

The wording of this paragraph has been revised to better convey the intended meaning.

Draft p. 3.1.74Issue

One commenter felt that the paragraph on "Candor" was an outrage. The large number of technical articles on waste disposal is not relevant. (40)

Response

The last sentence in the paragraph in question--"Some take this as evidence while others see the flood of articles as an attempt to confuse the layman and increase reliance on the technical expert." is a non-biased statement. It admits that the amount of published information on nuclear power may not be (in every instance) sufficient evidence of "Candor."

SOCIOECONOMIC/SOCIOPOLITICAL ISSUESDraft p. 3.1.75Issue

One commenter felt that the discussion of uncertainty should address the point that the magnitude of uncertainties alone can not determine whether one can proceed with any technological program. (40)

Response

This particular paragraph is under a discussion of "Non-technical Issues" and simply states some of the anomalies of the perception of many people towards uncertainty. There is no inference and certainly no statement in that paragraph which argues that because uncertainties are low one can proceed.

Draft p. 3.1.126-132, 3.1.179-181, 3.1.184-186, 3.1.193-195, 3.1.200Issue

GEIS is characterized as generic and not site specific (page 3.1.98). The document further states that the ability to identify socioeconomic impacts increases as one proceeds from a generic to a site-specific situation. However, a model was employed which provided and compared very specific social service demands anticipated for each of the reference sites. It is unclear why the analysis, which used actual site specific population, employment, education and housing information to estimate service demands, did not relate the demands to existing capacities to indicate net impacts.

The reference sites are compared and the comparison reveals a range of different conditions and anticipated social service demands. Are these reference sites being presented as being representative of sites to be found in the Southeast, Southwest and Midwest areas of the country? How much variability can one expect to find among sites within the geographical boundaries of each of the above areas (Southeast, Southwest and Midwest)? If large differences are expected within each of the geographical areas, to what use is the reviewer to put comparative information presented in GEIS?

While a considerable amount of useful information is presented in terms of manpower needs and expected social service demands for the three reference sites, the demands are not related to the infrastructure capacities of the expected impacted communities to ascertain net impacts. The subjects of compensation, payments in lieu of taxes, and mitigation in general, need considerably more development. (208-NRC)

Response

Population distributions chosen for analysis are reasonably representative of the range of distributions likely for siting repositories and are not representative of

SOCIOECONOMIC/SOCIOPOLITICAL ISSUES

specific areas per se. Large, qualitative variation of mechanisms to handle social service demands within each type of distribution would make such analysis unrepresentative and possibly misleading. Analyses suggested by this comment are meaningful when actual (not hypothetical) sites are considered in a specific EIS.

Draft p. 3.1.129

Issue

Social service demands (Table 3.1.21) were derived by applying factors "to the project in-migration values" (Table 3.1.19). Therefore, the level of forecasted social service demands by individual site should be proportional to the estimated level of project in-migrants for each site. From Table 3.1.19, under the maximum impact condition the respective estimates for the number of project in-migrants for 1985 indicate the lowest value for the Midwest site (5800), followed by the Southeast Site (8600) and the Southwest site (15,000). However, in draft Table 3.1.21, also under the maximum impact condition, some of the social services--physicians and dentists, and hospital and nursing care beds--indicate values for 1985 which reverse the relative position of the Midwest and Southeast sites. This apparent error occurs in similar tables throughout the DEIS. (113-EPA)

Response

Some social service demand factors are not constant across sites, since they are based upon observed levels of social service provision, usually for the state containing the site. For example, the expected demand for physicians per 1,000 people is 0.86 at the Southeast site and 1.33 at the Midwest site. This is based upon the number of active non-Federal physicians providing patient care in those areas in 1973. Each of the factors listed under the Health category are larger for the Midwest site than either of the other two sites because of the metropolitan character of the state containing this site. All other factors are the same for each site, except for the crime index which is also based on state-level data. In addition, demand for nursing care beds is based on that component of the in-migrant population aged 65 and over and is therefore subject to variation in population age composition as well as population size. Thus, whenever the in-migrant population of the Southeast Site exceeds the in-migrant population of the Midwest site by a small amount, health services likely to be required by the Midwest in-migrants may exceed those required by the Southeast in-migrants for the same time period.

Draft p. 3.1.129

Issue

The following statement appears to be incorrect in light of the information presented in the accompanying tables:

SOCIOECONOMIC/SOCIOPOLITICAL ISSUES

"Although the numbers of in-migrants are smaller, the potential for impacts in the Southeast maximum impact condition is quite similar to the potential in the Southwest site under maximum conditions. This is the case because the base population in the Southeast is roughly twice that in the Southwest site; therefore, the Southeast is capable of adsorbing greater population influx, other things being equal."

It appears that the words "Southeast" and "Southwest" should be reversed, since if the number of in-migrants for site A is half the number of site B, and the number of in-migrants stated as a percent of each site's base population is the same for each site, then the base population of site B must be twice the base population of site A.

The identical statement is repeated in other portions of the GEIS (pp. 3.1.120 and 3.1.194) in referring to other estimates of the numbers of in-migrants associated with different types of waste management facilities. (113-EPA)

Response

The words "Southeast" and "Southwest" have been inadvertently reversed. The referenced statement on pages 3.1.129, 3.1.180, and 3.1.194 of the draft (Section 5.4 of final) should be read as follows:

"Although the numbers of in-migrants are smaller, the potential for impacts in the Southeast maximum impact condition is quite similar to the potential in the Southwest site under maximum conditions. This is the case because the base population in the Southwest site is roughly twice that in the Southeast site; therefore, the Southwest is capable of absorbing greater population influx, other things being equal."

Because the Southwest regional population base is considerably smaller than at the other two sites, in-migration tends to be high under both expected and maximum impact conditions. The reason that in-migration is proportionately greater for the Southeast and Midwest sites between expected and maximum conditions is due largely to a greater increase in secondary employment multipliers for these two sites, and the assumption that fewer jobs will be filled by regional unemployment, thereby reducing the effect of large differences in regional population size. The corrected wording of the above paragraph should not diminish the conclusion that the Southwest site is most likely to experience the largest socio-economic impacts as defined in this analysis.

Issue

Several commenters noted that the Statement should expand on the discussion of non-technical (or sociopolitical) issues in the site selection process.

Non-technical issues will dominate the site selection process by being the initial barrier which must be overcome before further investigation is possible. Therefore, more detailed discussion of these various issues may be warranted to delineate all the possible options which are available to overcome these potential barriers. (7)

SOCIOECONOMIC/SOCIOPOLITICAL ISSUES

Expand the analysis of the "sociopolitical" factors which will determine access to any technically feasible site. (18)

Specify a firm mechanism for state and local participation prior to detailed site investigations in Stage III of the proposed site selection process. (43)

Response

The non-technical issues that are relevant to the site selection process have been presented in final Section 2.3 and Appendix B. Further discussion of non-technical issues along the lines suggested is not considered appropriate for a generic impact statement because such considerations will be highly site-specific and cannot be detailed at this time.

Issue

Several commenters pointed out that the draft statement did not sufficiently address institutional issues relating to waste management. (35, 38, 154, 198)

Response

As a result of the revised structure (outline) for Volume 1, information relevant to nontechnical issues that did appear in several places of Section 3.1 of the draft was drawn together and presented in a single section (final Section 3.5). The function of this discussion is to air non-technical issues relating to nuclear waste management. In addition, the section on Technology Comparisons (final Section 6.2) uses Domestic Political Considerations and conformance with Federal Law International Agreements as two of the factors on which the disposal options are examined and evaluated.

Issue

One commenter stated that the draft Statement did not adequately address the uncertainties and risks with regard to future institutions and how this relates to the likelihood of securing a repository from man-caused events. (114)

Response

This Statement does address the issue of human institutions in long-term waste management. It notes that there is debate over the roles human institutions may have on long-term management of nuclear wastes including the following:

1. The functions that can or should be performed.
2. The subjective need for these institutions.
3. The likelihood the functions will be performed at any point in time.

A supporting document (Hebert et al. 1978) outlines in more detail non-technical issues in nuclear waste management and serves as the basis for the discussion in Section 3.5.

SOCIOECONOMIC/SOCIOPOLITICAL ISSUES

In addition, final Section 5.5 analyzes two man-induced repository breach scenarios (repository breach by drilling and, in the case of salt, solution mining). The consequences of these events are presented; however, no attempt is made to integrate the probability of occurrence into the presentation.

Issue

A question was raised as to how the issue of future institutions is to be considered in the overall decision-making process. (114)

Response

Upon implementing a specific disposal strategy, the area of institutional control of repositories will be further investigated.

Issue

One commenter stated that the future economic and employment impacts of commercial nuclear waste management should be considered. (15)

Response

These factors were considered in the analysis for entire waste management systems (see Sections 4.7 and 5.4 and Chapter 7.0).

Issue

Analysis of both the microeconomic and macroeconomic impacts should be performed. Within the micro framework, the direct impacts on the customer's electric rates and fuel bills should be investigated. Macroeconomic considerations should include the degree of secondary impacts stemming from a rate increase to commercial and industrial electric users which can influence the cost of producing other goods and services in the economy. The economic impacts of the cost of waste management also need to be discussed on a regional basis since they depend on each area's relative reliance on nuclear-generated electricity. (113-EPA)

Response

The economic impact for the consumer has been addressed at least to the extent of determining the mils/kW-hr that would no doubt be added to the rate payer's charge because of waste management activities. Going beyond that point in this generic Statement was not believed to be warranted.

SOCIOECONOMIC/SOCIOPOLITICAL ISSUESIssue

One commenter suggested that the Statement should emphasize the importance to the economy of proceeding with geologic disposal. (20)

Response

As noted above, economical employment impacts directly attributable to waste management activities were investigated. Other macroeconomic concerns (as perhaps suggested by the commenter) were not examined.

Issue

DOE's treatment of "socioeconomic impacts" seems designed to give more weight to plans for placing radioactive waste repositories in already populated areas where there would be a large employee population force in place. This is a specious consideration when it is made without consideration of the very considerable environmental and safety impacts being imposed on large segments of the population. (55)

Response

The Statement examines three reference environments with differing demographic characteristics for the purposes of analyzing socioeconomic impacts of repository construction and operation. For details of this analysis see final Sections 3.2, 4.7 and 5.4 and Appendix G.

Issue

One commenter felt the the analysis of socio-economic impacts should be modified. The analysis suffers from:

- failure to treat the socio-economic and institutional impacts of a stable nuclear economy;
 - sensitivity of estimates of effected to choice of reference site;
 - inappropriate application of the analysis across various disposal options;
 - overly simplistic indicators for assessing complex social and economic impacts;
 - incomplete analysis of impacts associated with repository decommissioning; and
 - superficial discussion of equity issues and possibilities of their resolution.
- (219)

Response

DOE would suggest that there is a need to strike a balance between what is useful to the reader in a generic analysis versus the more complete analysis that might be required in

RESOURCE REQUIREMENTS

a site specific situation. As noted throughout this Section, DOE has attempted to improve the presentation of socioeconomic impacts and institutional issues relating to waste management.

Draft p. 1.17

Issue

One commenter requested that more detailed information should be provided as to how the energy needs vary across fuel cycles and geologic media. (34)

Response

This information is presented in Chapter 7.0 and in the support document DOE/ET-0029 (Chapter 2.0).

Draft p. 3.1.116

Issue

Table 3.1.11 purports to give estimates of resources needed for construction of operation of waste repositories in various geologic formation for different fuel cycle options. It also compares effluents for the various options. However, no basis for any of the numbers listed is given. The basis for such estimates should be included. (208-NRC)

Response

The sources of the information in draft Table 3.1.11 are DOE/ET-0028 and DOE/ET-0029.

Draft p. 3.1.116

Issue

One commenter requested that the values in Table 3.1.11 be denormalized. What is the daily electrical use, and what percentage of the model site is this? (43)

Response

Denormalized values can be found in DOE/ET-0029. The tabulations referred to have been put in terms of a model facility in the final Statement (see Sections 4.7, 5.4, and 5.8)

RESOURCE REQUIREMENTSDraft pp. 3.1.179, 183, 188, 203Issue

No basis for any of the values of resources committed shown in Tables 3.1.55, 62, 65, and 78 are given. In addition, no units are given for operational water use, concrete, propane and electricity in draft Table 3.1.55. (208-NRC)

Response

See DOE/ET-0029 for basis of values in draft Tables 3.1.55, 62, 65 and 78. Units for operational use of water (m³), concrete (MT), propane (m³) and electricity (kW) were missing from draft Table 3.1.55. These omissions have been corrected.

Draft p. 3.1.189Issue

A question was raised as to how much energy would be required for the surveillance of nuclear wastes following disposal. (30)

Response

DOE anticipates some type of post-closure monitoring system will be implemented for as long as future generations care to operate it. While the amount of energy expended would be dependent on the length and extent of the monitoring effort, it is anticipated that the energy requirements would be less than that needed for the operation of the various waste management facilities (equivalent of 0.08% of energy produced in power plants through the year 2050--250 GWe growth and decline scenario).

Draft p. 3.1.223Issue

The third paragraph suggests that water use will not be a problem. The basis for the statement was the assumption that the facilities could all be located near the "R" River, which had adequate flow. However, the statement should recognize that water use could be a significant environmental impact for a repository which cannot be located near a convenient water source.

The resource commitments listed include annual water use for the once-through fuel cycle option. The total annual use is about 1% of the annual mean flow of the "R" River, a small amount when water is plentiful. However, in the semi-arid west where river flows can be less than 100 cfs (one-fiftieth that of the R River) and where water is fully allocated, this is a significant amount of surface water use. (208-NRC)

REFERENCE ENVIRONMENTS

Response

DOE agrees. Care must be taken when siting a repository. The selection of such sites will be based on several site-specific parameters.

Issue

Several letters commented on the reference environment approach.

Draft Appendix F--The material presented on the reference environment was abbreviated and generally inadequate. (12)

Draft Appendix F--Why are additional reference environments not considered? If the reference environment described is located in Wisconsin, then serious consideration would have to be given to transportation problems. (43)

Draft Appendix F--Failure to identify the location of the reference environment has resulted in fear that DOE will not seek input from local officials when siting a repository. (129)

Location of the generic site in the middle west typifies the industry's disregard for human life. (96)

Inasmuch as the GEIS is a programmatic statement, a site-specific description of an environment is not necessary; however, development of data that will be required in a specific evaluation is appropriate, and the GEIS incorporates a reference environment to evaluate source terms on a generic basis. However, once having determined the significance of an impact on the reference environment, the GEIS fails to remind the reader that conclusions reached relate only to those particular conditions. Indeed, statements in the GEIS indicate that even its writers do not fully appreciate these limitations. Effects on the reference environment are presented as the impacts of an alternative without recognition of the fact that the impacts could be much different for a different reference environment. For an example of how to prepare a GEIS with detailed discussions of siting options and impacts, see the FES on Floating Nuclear Plants (NUREG-0056). (208-NRC)

Response

The reference environment concept was originally used to deal with the potential ecological and other impacts of geologic disposal, and the environment was representative of the North Central United States in a moderately wet environment. Subsequently some comparisons were made to an arid Southwestern U.S. location (draft Appendix G also compares socioeconomic aspects of these sites). To try to deal with ecological impacts on a generic basis is not very meaningful; the adoption of the reference environment was used to provide a degree of site specificity in order to overcome this problem.

REFERENCE ENVIRONMENTS

The complaint that the Statement fails to remind the reader that conclusions reached relate only to the particular conditions set forth in the reference environment are perhaps valid. To respond to this comment the following addition was made to Section 5.4 of the final Statement. "The reader should keep in mind that impact assessments and comparisons in this report that are based on reference environments are to a degree site specific and therefore limited to one set of environmental conditions. The conclusions reached relate only to these particular conditions and are not applicable in a generic sense."

The reference environment used as a basis for assessing environmental impacts of construction and operation of waste treatment, interim storage, and/or final disposal facilities (draft Appendix F) was developed, with changes, from information presented in the final Environmental Impact Statement on operation of the Monticello (Minnesota) Nuclear Generating Plant (Docket Number 50-263, November 1972). The reference demographic environments (draft Appendix G) used for determining socioeconomic environments impacts were centered around Monticello, Minnesota (Midwestern U.S.); Barnwell, South Carolina (Southwestern U.S.); and Eddy County, New Mexico (Southwestern U.S.).

The data base for the four geologic media analyzed (salt, basalt, granite, shale) was developed from a literature search after which real stratigraphic sections were compared and combined into a composite section generally representative of each study area. By subjectively comparing and combining the composite stratigraphic sections, the generic stratigraphic section was developed for each media type. In developing the generic stratigraphic section for salt, the properties of bedded salt deposits from three different areas (Salina basin, Permian basin, Paradox basin) were considered. Gulf coast salt domes were also investigated. The generic basalt stratigraphy was developed after reviewing rock property data for the Dresser, Amchitka, Nevada Test Site and Columbia River Group basalts. The "typical" shales selected for review were the Pierre, Chagrin, and the Waynesville and Arnheim formations. The granite formations selected for study were the Barr, Colville, Pikes Peak, and Charcoal granites.

Use of these reference environments and geologic media characterizations is not to be construed as a selection of these areas, types of environments, or media characterization for any nuclear fuel cycle facility. More specific environments will be addressed in later environmental impact statements.

Draft p. F.2

Issue

The description of the geology at the bottom of page F.2 would be improved if there were some indication of the depth of the basement rocks and of the general nature of the overlying rocks. (113-EPA)

REFERENCE ENVIRONMENTSResponse

The reference environment was developed principally for above ground environmental analyses such as population dose which requires a description of demography and was not intended to be considered of as a candidate site.

Draft Appendix FIssue

If the maximum flood on record was 3 meters above the normal river stage, and the one in a thousand year flood would be expected to be 5 meters above normal river stage, under what conditions would the "maximum probable flood" which is 10 meters above normal river stage be expected. Is this a once in a million year flood? (113-EPA)

Response

The probable maximum flood is the worst case flood. No time is specified.

Draft Appendix FIssue

There is a short paragraph on ground water, but nothing as to the nature of the aquifer--permeability, hydraulic gradients, or retardation factors. This should be included. (113-EPA)

Response

This type of information was not available in sufficient detail to be useful. Moreover near surface features were considered adequate for the use to which the information was put.

Draft p. F.3Issue

The hydrology of the hypothetical site is presented with no explanation or discussion of its appropriateness for general sites. No discussion of other hydrologies is given. Considering the great length of discussion that is given throughout the document to effects of comparatively small changes in the characteristics of the waste, an apparent lack of appreciation of the effects of the sites hydrologic characteristics is manifested by this treatment. (208-NRC)

REFERENCE ENVIRONMENTSResponse

The problem is that no generic site can be justified. The earth's aquifers have a broad range of measurable characteristics. DOE cannot define a generic groundwater system, because there will always be disputes over the definition. The best that can be done is to use a conservative approach.

COMPARATIVE ASSESSMENT

Draft pp. 1.1, 31, 35, 36

Issue

The GEIS is self-contradictory on whether it is recommending a particular decision or decisions. In some sections it appears a certain course of action is being recommended. In particular on draft p. 1.36, after eliminating most other factors as unimportant, it is stated, "Thus, state of technology stands out as a major decision factor, and the geologic disposal option has an edge over other options as regards the technology status." On draft p. 1.1 it is stated: "DOE proposes that 1) disposal of radioactive wastes in geological formations can likely be developed and applied with minimum environmental consequences, and 2) therefore the program emphasis should be on the establishment of mined repositories as the operative disposal technology."

However, as indicated on draft p. 1.31, the comparative analysis is intentionally not completed to "avoid value assumptions--more appropriately the responsibility of the decision maker." On draft p. 1.35 is found: "It is emphasized that the scores in draft Table 1.8 cannot be combined without careful consideration of the relative importance of the attributes and of the criteria." The relative importance was not determined. Further, draft p. 4.1 states that "No attempt is made to identify specific CWM options for further research and development." Draft p. 4.24 reiterates that weighting factors have not been assigned and decisions not recommended.

The GEIS should not terminate the comparative analysis midway before assigning weighting factors, disclaim the making of a recommendation, and then proceed to make such recommendations as are found on pp. 1.36 and 1.1. (208-NRC)

Response

The statements quoted from the draft document in the first paragraph are correct. The contradictions (and inconsistencies) referred to in paragraph two were corrected and the comparative analysis has been revised to specifically identify and rank those technologies warranting continued development.

Issue

Several commenters felt that the comparative assessment should more clearly emphasize differences between disposal alternatives and identify to what extent DOE's program will pursue disposal options other than geologic disposal.

COMPARATIVE ASSESSMENT

Draft p. 1.3--The final Statement should eliminate from further consideration most if not all of the alternatives other than geologic disposal. (154)

Draft p. 1.23-1.30--The Statement should enable the reader to discern what alternatives (aside from geologic disposal) should receive further program emphasis. (154)

Draft p. 1.31-36--The summary comparative analysis should concentrate on the recommendations for the entire program and note which alternatives (in addition to geologic disposal) should receive further study by DOE. (208-NRC)

Draft p. 1.35--The conclusion that state of technology is a major decision factor and that geologic disposal has an edge over other options in this regard should be more explicitly supported in Chapter 4.0. (198)

Draft p. 4.1--The final Statement ought to provide to the decision maker a reasonable summary of the bases upon which the identification of specific options for further R&D can be made. (154)

The information provided on the ten alternative concepts shows clear and significant advantages for three of the concepts: geologic disposal, island disposal, and shale grout injection. Chapter 4 fails to bring out this distinction. (11)

The final Statement should improve the comparative assessment as much as possible as should specify to what extent the program will pursue alternatives other than geologic disposal. (113-EPA)

The Statement should address the feasibility of, the level of R&D funding and the basis upon which the alternative disposal concepts should be reconsidered as the basis for a future proposed Federal action. (154)

Some method of emphasizing differences between the disposal alternatives should be sought. (201)

The need for continued pursuit of other technologies should be more heavily stressed in the final Statement. (218-DOI)

Criteria could be used to more effectively present the advantages, disadvantages, and unresolved technical, sociological, political, and esthetic issues involved with various disposal options. (218-DOI)

Response

The section on Comparison of Disposal Technologies (final Section 6.2) was revised with the following objectives:

1. To use criteria that are both relevant to an environmental impact statement and would assist the decision-maker in distinguishing between the disposal technologies.

COMPARATIVE ASSESSMENT

2. To structure the section in order to specifically identify and rank disposal options warranting continued development.

This revision is also reflected in the Summary Chapter (1.0).

Draft p. 1.31

Issue

In comparing the options the claim is made that "value judgements are not within the scope of this document." An EIS should provide at least the basis for value judgments in our opinion. (154)

Response

When examining the alternative disposal options, the basis for comparisons made are provided. See final Section 6.2.

Draft p. 1.31

Issue

Several commenters noted that the names and qualifications of the people who comprised the "panel of Experts" involved with the comparative assessment of alternatives should be discussed. (208-NRC, 217)

Response

Quantitative assessments (utilizing the consensus of a panel) were replaced by more qualitative interpretations of information by the entire DOE research team involved in preparation of the final Statement. A presentation of the Department's authority and experience in the waste management area can be found in DOE's Position Paper to the NRC rule-making proceedings (DOE 1980a).

Issue

Several letters commented on the numerical rating scheme used in the draft Statement.

Draft p. 1.31--The significance of the comparative analysis is clouded by the use of scales that are nonlinear with no relative scaling distributions given and nonindicative of acceptability (e.g., page 4.10 contains a statement that "...'five' the maximum rating does not necessarily represent a 'good' situation..."). (208-NRC)

Draft pp. 1.34 and 4.2--Most of the rankings in Table 1.8 are value judgments in spite of the fact that it stated that the matrix approach was used to minimize value judgments. It is inconsistent to make estimates for some criteria and not for others. (218-DO1)

COMPARATIVE ASSESSMENT

Draft p. 4.10--The statement "Strict scale linearity should not be assumed" contradicts earlier statements concerning the absolute nature of the intervals. (147)

Draft pp. 4.10-11--We seriously quarrel with the analysis. It is inconceivable that several of the alternatives rate a score as close to geologic disposal as they do. We hope the final EIS present a summary which is not only more realistic but of more use to the decision maker. (154)

Draft p. 4.11--Qualitative numbers are assigned in Table 4.5.1 on the basis of subjective judgment in areas where technology is admittedly thin. Such an analysis should be deleted. (11)

Response

The numerical scale used in the draft was replaced by a qualitative method of examining the disposal concepts. The final Statement (see Section 6.2) also includes a discussion of the performance objectives outlined in the recent DOE Position Paper to the NRC confidence rulemaking proceedings on nuclear waste storage and disposal (DOE 1980a) and addresses the degree to which the disposal technologies meet these performance objectives.

Issue

Many letters noted that the comparison of alternatives did not give sufficient consideration to environmental factors.

Draft p. 1.34, Table 1.8--It is strongly disagreed that insufficient data exists to determine ecosystem impacts. (147)

Draft p. 1.35--We were disappointed that the body of the Statement seemed to have considerable information on ecosystem impact yet the Summary concludes that "Insufficient data were available to evaluate Ecosystem Impact criterion." (34)

Draft p. 1.35--It is difficult to make a decision without complete data on ecosystem impacts. The comparative assessment fails to present reliable data on ecosystem impacts. (128)

Draft p. 4.11--Table 4.5.1 indicates that insufficient data is available to compare ecosystem, aesthetic, and critical resource consumption impacts. These are among the most basic and fundamental, true environmental impacts. The majority of the remaining criteria are better described as policy considerations than as environmental factors, e.g., status of technology, cost of construction, policy and equity considerations. Thus, it appears that the final comparative analysis in this environmental impact statement drops out environmental factors and is based on the policy considerations. Environmental impacts, other than dose assessments, such as hydrologic impacts including water use and availability impacts of construction and operation of the repository need more detailed discussion. The GEIS does not present sufficient information on ecosystem impacts or critical resource consumption impacts. (208-NRC)

COMPARATIVE ASSESSMENT

Draft p. 4.22--The final Statement should identify what the potential ecosystem impacts are with the management of commercially generated radioactive waste and consider the radiological effects on plants and animals in addition to humans. (218-DOI)

The comparative assessment fails to provide reliable data on ecosystem impacts. (167)

The GEIS does not present sufficient information on ecosystem impacts or critical resource consumption impacts. (217)

Response

The most complete environmental assessment was made for the geologic repositories, and for salt in particular. To provide what is asked for (the provision of more detailed information and a more detailed discussion/comparison of the environmental impacts of alternatives) would require a considerable effort in acquiring site specific data. For ecological impacts, the site specific kinds of baseline information needed for comparing alternative disposal methods may be lacking; and the generic treatment of comparative impacts may not have much meaning unless one were to develop a set of conditions (i.e., a reference environment) for each alternative. The final Statement includes updates of information available concerning alternatives. However, much of the information remains site specific and is not included.

Draft p. 1.34, Table 1.8Issue

One commenter stated a criteria reflecting internal policy conflicts be included. (218-DOI)

Response

Section 6.2 of the final Statement includes a criteria called Domestic Political Considerations.

Draft p. 1.34 Table 1.8Issue

The section entitled Socio-economic Impact mentions that the impacts were not converted to a 1 to 5 scale; refers the reader to Table 4.5.2 and states that the impacts are small for all options. It would be helpful to discuss why the 1 to 5 scale was not used and what rationale was used in both tables to conclude that the impacts would be small. These conclusions appear to be at variance with the statement made on page 1.22 (line 19) which states: ". . . socioeconomic impacts . . . could be either small or significant."

COMPARATIVE ASSESSMENT

Also refer to the statement on page 3.1.47 (lines 10 and 11) which points out that socioeconomic and political factors may eventually play a determining part in repository site selection. (208-NRC)

Response

This comment is well taken and has been incorporated in the revision of the draft (Section 6.2.4.2).

Draft p. 1.36, 4.11

Issue

"Years until operational" is picked as the major decision factor in selecting technology. But, a basis for considering this to be an important factor, that is a near-term need is not articulated. On page 5.1, it is indicated that alternatives have been ranked with respect to the ease and likelihood of implementation by "the design target date" to evaluate development status of technology. What this target date is is not revealed. This approach is backwards in any event as the GEIS should present information to support the determination of a need date or of need as a function of time and not evaluate options by assuming a need date. (208-NRC)

Response

The DOE Position Paper to the NRC rulemaking proceedings (DOE 1980a) notes as one of its objectives for safe and environmentally acceptable disposal of high-level waste that "waste disposal systems selected for implementation should be based on a level of technology that can be implemented within a reasonable period of time, not depend upon scientific breakthroughs, should be able to be assessed with current capabilities . . .". The DOE Position Paper also stated a range of target dates for the availability of the first mined repository (1997-2006).

Draft p. 3.1.246

Issue

In the last paragraph on draft page 3.1.246, it is stated that "Table 3.1.95 presents for conventional geologic disposal the data used as a basis for scalar quantities in the comparative analysis discussion." Table 3.1.95 implies that there is "no data" in a number of key areas for making a comparative analysis. Based on this, it would appear that 1) no substantive basis exists for making a rational comparison among disposal options and 2) there may not even be a sufficient basis for assessing the expected environmental impacts from conventional geological disposal. (208-NRC)

COMPARATIVE ASSESSMENTResponse

The reference material was not well worded and modifications were made. Moreover it is believed that the data base exists to generically assess the adequacy of geologic disposal.

Draft pp. 4.1-45Issue

Chapter 4.0 gives little guidance in judging the relative environmental and social impacts of the possible courses of action. (208-NRC)

Response

The revised comparative assessment attempts to address the potentially significant environmental and social impacts of the waste disposal alternatives to a depth appropriate for this assessment and current level of knowledge.

The President, in his February 12, 1980, message, noted that past governmental efforts to manage radioactive wastes have neither been technically adequate, nor have they sufficiently involved states, local governments and the public in policy and program decisions. This message established a program with mechanisms for full participation of these groups and continuous public review. The Department of Energy is fully committed to this program.

Draft p. 4.2Issue

There seems to be a contradiction between the statement on page 4.2, second paragraph, which says: "Value judgments were required in at least two areas: 1) judgments relative to selection of the decision criteria and 2) judgments relative to selection of appropriate methods of measuring effects on criteria," and the statement in the footnote on p. 4.2 which says: "Because these questions relate to the values of society and individuals they are avoided here where possible." (208-NRC)

Response

It is stated that "...they be avoided here where possible." In this instance, value judgments were necessary and their avoidance was not possible.

COMPARATIVE ASSESSMENTDraft p. 4.4Issue

Table 4.2.1 indicates that "non-high-level" TRU wastes cannot be disposed of by, among others, the very-deep hole, island disposal, and subseabed disposal methods. It is not apparent why this is so. The GEIS should either present a rationale for requiring separate disposal methods or include "non-high-level" wastes in the wastes to be disposed of by those disposal methods. This is important because the current GEIS assumptions require that if disposal of HLW by the above methods is used, disposal in mined cavities in bedded salt also be an acceptable method. (208-NRC)

Response

Non-high-level TRU wastes can be disposed of by methods other than geologic disposal; however, other constraints, principally volume, make geologic disposal of these wastes the preferred alternative. This distinction was brought out in Section 6.2.

Draft p. 4.7Issue

Beginning on page 4.7 eleven decision criteria are presented and discussed. One is called Ecosystem Impact and consists of two attributes. No rationale is given for selecting these particular measures as criteria. On p. 4.11, Table 4.5.1 states that available information on the physical and operating characteristics of the commercial waste management options is not sufficient to permit comparative assessment of these attributes. Appendix F does not give any primary production information. While Table 3.1.95 presents data used as a basis for scalar quantities in comparative analysis. They give a value of 5×10^{10} g dry organic matter for reversible ecological effects. There is no explanation of where this number comes from or why it is used except that on page 5.19 a formula is given for determining primary production. (208-NRC)

Response

The attributes cited under the Ecosystem Impact criterion in the draft Statement were removed during preparation of the final document. The data relating to reversible ecological effects was also deleted from the Statement.

COMPARATIVE ASSESSMENTDraft p. 4.9Issue

Determining net primary production has no value in deciding which option should be selected nor in making decisions at other levels in the program. (208-NRC)

Response

The "Net Primary Production" measure was removed from the final Statement.

Draft p. 4.16Issue

One commenter noted that discussion of the criterion Safeguards and Security is incomplete for several reasons. The GEIS has been prepared for decision makers and the public. The uranium-only recycle has not been addressed in this draft GEIS from a safeguards standpoint. This and other cycles could have significant safeguards implications. In addition, this section attempts to identify the purpose of proposed safeguards systems but does not provide the decision maker or public with a discussion of the concept or elements of proposed systems for specific forms of waste. It will be difficult to form a judgment on the adequacy of any safeguards system without this information. (208-NRC)

Response

The uranium-only cycle has been deleted from consideration in the final Statement. An expanded discussion of safeguards considerations for other waste forms is included in the final Statement (see Sections 4.10, 5.7, and 3.2.9).

Draft p. 4.16Issue

One commenter noted that the footnote at the bottom of p. 4.16 is not accurate. The Nuclear Regulatory Commission is studying this problem but has not yet published safeguards requirements specifically applicable to waste repositories. (208-NRC)

Response

This has been deleted from the final Statement.

COMPARATIVE ASSESSMENTDraft p. 4.44Issue

There are references to: "some argue that public confidence would be lost..." and on the first paragraph, page 4.45: "some people argue that..." Are these people DOE staff, results of public survey, comment letters? Who "some people" are should be specified. (208-NRC)

Response

The material in question has been deleted from the final Statement.

Draft p. S.1Issue

Specialties of experts that assessed a number of effects are given but it is not stated what the specialties of experts that assessed ecosystem impacts were. (208-NRC)

Response

Specialties of the experts assessing ecosystem impacts were included in the draft where applicable.

Draft p. S.3Issue

One commenter noted that the uranium-only cycle should be included in the discussion and factors of attractiveness should be identified for this cycle. Because of the presence of plutonium in this cycle the sabotage and the theft susceptibilities should be analyzed separately.

Consequences and environmental impacts of successful acts of dispersal, sabotage or theft have not been considered in establishing the susceptibility index. These factors could have a bearing on the level of safeguards required in factor number 3 in the short-term susceptibility to encroachment case.

The level of safeguards appropriate for a type of waste appear to be based upon an evaluation concerning the types of wastes which would be attractive for theft or sabotage. This attractiveness criterion is inherently conjectural and should not be used as a basis for determining safeguards requirements. The appropriate considerations in this area are the potential consequences to public health and safety and common defense and security that result from successful theft or sabotage of each specific type of waste. (208-NRC)

COMPARATIVE ASSESSMENTResponse

See response to issue in this section referring to p. 4.16. Also, the relationships between environmental consequences and risk is covered in Section 3.2.9 of the final Statement.

Draft p. S.11Issue

One commenter noted that Table S.3 "Proposed Safeguards Requirements" does not include any material control and accounting requirements. Safeguards requirements for a high-level waste repository might include some form of accountability requirements during the period prior to final closure, particularly in the case of the uranium-only fuel cycle where significant quantities of plutonium would be present. (208-NRC)

Response

Material control and accounting requirements are discussed in Sections 3.2.9 and 5.7.2 of the final Statement.

Draft p. S.19Issue

Under Ecosystem Impact, it is stated that the significant ecological effects may occur from construction of buildings, etc. No basis is given for this conclusion. (208-NRC)

Response

The conclusion that ecological effects may accrue from construction of facilities and the establishment of exclusion areas is based principally on expected changes in land use and the associated modification of plant and animal habitat. As mentioned in the paragraph in question, these changes include paving and construction of retention ponds and spoil piles. The effect of change will depend on site-specific parameters. The value of the natural and managed resources at risk, the uniqueness of the affected habitat and the presence of rare species are all factors to be considered. Construction impacts may be relatively short-term and reversible, lasting only during the construction period or through the life of the facility.

Issue

Criteria for which sufficient data not available should not be included in the comparative assessment. (217)

COMPARATIVE ASSESSMENTResponse

One of the benefits of a structured decision analysis is the identification of important issues for which additional information should be obtained. Consequently an attempt was made to include all major issues of importance in the comparative analysis.

Issue

One commenter suggested that the Statement discuss the concept of mitigation in terms of taking corrective action if unanticipated events occur. (219)

Response

The concept of mitigation was considered as a criterion in the final Statement (see Section 6.2).

Issue

One commenter felt that the comparative assessment was not valid and that the exercise should be performed over. (217)

Response

DOE agrees with the commenter and has performed the exercise over (see Section 6.2).

Issue

Several commenters suggested that the Statement present an integrated assessment of the disposal alternatives. (38, 58)

Response

The draft Statement did evaluate each of the disposal options from a systems perspective (e.g. examination of the seabed concept requires consideration of treatment, storage, transportation, and disposal operations for the entire waste stream). In preparing the final Statement the latest data available was utilized in developing this comparative assessment.

Issue

One commenter noted that all of the alternatives to conventional geological disposal recommend separate disposal of non-high-level wastes in mined salt repositories. The lack of an alternative to salt repositories for this waste is noteworthy, unacceptable, and should not be allowed to bias overall choices for the waste management program. (28)

COMPARATIVE ASSESSMENTResponse

The choice of salt (in the draft Statement) was for the purpose of simplifying comparisons. It does not imply that non-high level wastes could not or would not be placed in repositories of other rock types. The selection of other rock types would not alter the conclusions of the comparative assessment. References to salt as a medium for disposal of non-high level waste for those alternatives requiring ancillary non-high level waste disposal were deleted from the final Statement.

ALTERNATIVE DISPOSAL CONCEPTSGeneral

Draft p. 1.23, 24, 25, 27

Issue

One commenter noted that the nature and relative importance of research (necessary for the alternative concepts) should be expressed more clearly in the final EIS. (154)

Response

In preparing the final Statement, effort was made to present the research and development requirements in a clear and concise discussion for each of the concepts (see Sections 6.1.1.3, 6.1.2.3, . . . 6.1.8.3).

Draft pp. 3.1.136, 3.3.3

Issue

Where there exist areas of uncertainty common to different alternatives they should be equally treated. For example on page 3.3.3 it states, "Information to satisfactorily assess the feasibility of the very deep hole concept is inadequate. This is not to say that the concept is not feasible, but there is not sufficient knowledge at present to confirm that radioactive waste can be isolated deep enough. . .to avoid transport of radioactive material to the biosphere. The main uncertainty is the lack of information about porosity, permeability and water conditions at great depths." On page 3.3.1 of the GEIS it states that very deep hole disposal is considered flawed because more information is needed on groundwater systems, rock strength and sealing of holes over long periods of time. On the other hand it is argued on page 3.1.136 that no long term significant impacts are expected to result from waste repositories described previously in this statement whether located in salt, granite, shale or basalt formation. It would appear the information needs stated for deep hole disposal would also exist for conventional geological disposal. (208-NRC)

The only alternative that is covered in any degree of detail is deep geologic disposal. While it is realized that less information is available for other alternatives, it appears they could be considered in more detail than these have been. (208-NRC)

Response

The first comment correctly addresses the issue of uncertainty regarding knowledge pertaining to the hydrology of deep geologic systems. Current limitations of such knowledge are important reasons for not considering the very deep hole concept as the primary technical alternative. Also, there are other areas of uncertainty regarding the very deep hole concept that are important though not addressed by the commenter. These are addressed in the final Statement and include deep bore hole sealing, deep emplacement technology,

ALTERNATIVE DISPOSAL CONCEPTS

geochemistry at depth, deep rock mechanical characteristics, and temperature effects that would result from waste emplacement. On the other hand, extensive experience gained--for example, through natural resource exploration and development, including mining, and a variety of geologic exploration programs for other purposes--has provided a plethora of information relevant to hydrologic and geologic characteristics of mined repositories at depths between 500 m and 1500 m. Although this information is available and there have been many studies for mined repositories, information for depths beyond 1500 m becomes significantly less available as depth increases.

The second comment suggests that more detail should be considered in the discussion of impacts that may be attributable to alternatives. Such detail is included to the extent practicable in Section 6.1 of the final Statement.

Although the comments address two different subject areas, an important relationship is noteworthy; namely, both comments illustrate the significantly different levels of information that exist for mined repositories as compared to all of the other alternatives being considered. The mined repository concept has been the subject of a large number of studies over many years. The studies include transportation logistics, detailed repository design, waste and host rock interaction, to name a few general categories. The most advanced of the alternative concepts, subseabed, has only recently advanced to the stage where host media and waste material interactions are being studied.

Discussions presented in Section 6.2 and Chapter 7.0 of the final Statement address the issue of relative stage of development and its importance to the evaluation of alternatives and to the selection of a programmatic option for disposal of radioactive waste.

Draft p. 3.2

Issue

This entire section appears to be largely speculative. The comment as to the speculative nature of the discussion applies to Sections 3.2, 3.3, 3.4, and, in fact, all of the rest of Chapter 3.0. (113-EPA)

Response

Alternative concepts for the disposal of radioactive waste are at various levels of development with the mined repository being the most developed. For most of the concepts there are large uncertainties regarding their characteristics and attributes. These uncertainties are discussed in Section 6.1 of the final Statement. Section 6.2 presents a comparative assessment of alternatives in which the current status of development, and therefore uncertainty, is discussed. These sections also offer the reviewer a synopsis of a number of the various alternatives that have been proposed for the disposal of radioactive waste and a summary of the rationale for selection of the mined repository as the

ALTERNATIVE DISPOSAL CONCEPTS

the preferred alternative. In general, alternatives that are considered to have large performance uncertainties are ranked lower in preference than those having more well defined and acceptable performance characteristics.

Draft Section 3Issue

It would be useful to provide concise summaries of advantages and disadvantages for the three subsections now lacking them: chemical resynthesis (Section 3.2), very deep hole concept (Section 3.3), and space disposal (Section 3.10). (218-DOI)

Response

Section 6.1 presents a discussion of the attributes, uncertainties, disadvantages, and other characteristics of the alternative concepts. Section 6.2 of the final Statement provides a comparative assessment of alternative concepts. This assessment identifies the relative merits of the concepts under review consideration.

Issue

One commenter noted that the alternative disposal options should be given emphasis appropriate to their plausibility. (219)

Response

The final Section 6.1 on the alternative disposal technologies was revised to reflect the depth of information available. An assessment of the alternatives that provides a comparison of attributes, including current development status, is presented in Section 6.2 of the final Statement.

Issue

Sections of the EIS discussing the alternative disposal concepts do not discuss either the problems or costs associated with the eventual decommissioning of waste disposal sites. Is this to be our legacy to future generations? Can we not consider such implications now before passing on such an irrevocable "gift"? (167)

Response

Decommissioning costs, where information is available from the literature, are discussed in the final Statement. Section 6.1.1.6 is one specific example where the cost of decommissioning and decontamination are addressed, and Section 6.1.8.6 is another.

In the preparation of the final Statement the DOE has made efforts to assemble and use all of the relevant information on alternatives that is in existence. It is important,

ALTERNATIVE DISPOSAL CONCEPTS

however, to recognize that many of the alternative concepts have not been advanced to the stage where detailed design studies, operations analyses, and subsequent decommissioning costs have been prepared. Conversely, decommissioning of the mined repository concept (Chapter 5.0) has been studied and cost estimates for such decommissioning exist.

Issue

Several letters requested additional analysis of transportation operations.

The feasibility of accomplishing the required transportation as well as the impacts associated with them should be studied and presented as part of the development of the alternative concepts. (43, 97)

Transportation impacts vary widely among alternatives yet generally are dismissed, without much discussion, as being insignificant. (208-NRC)

Response

Greater attention is given to this subject in the final Statement. Section 4.5 discusses in detail waste transportation systems for mined repository disposal as well as alternative concepts. Sections 4.7, 4.8, 4.9 and 4.10 present the environmental impacts, accident analyses, cost analyses, and safeguard requirements for transportation operations (as well as other predisposal activities). The relationship between transportation activities and the alternative disposal techniques discussed in Section 6.1 is presented in Section 4.2 of the final Statement.

Issue

What are the difficulties associated with following disposal site means:

- very deep hole?--cost?
- rock melting concept?--contamination?
- island disposal?--security?
- subseabed?--contamination?
- icesheet disposal?--melting?
- reverse well disposal?--leakage?
- space disposal?--space probe collision?--contamination of the solar furnace? (82)

Response

With the exception of the space disposal issues cited, each of the commenter's questions is addressed in Section 6.1 of the final Statement. A complete listing of research reports pertinent to the alternative concepts is presented in Appendix M.

Space probe collisions have not been specifically addressed because such occurrences are thought to be extremely unlikely. However, should the space disposal concept be pursued, careful analysis of system operations, system failures, accidents, and interaction

ALTERNATIVE DISPOSAL CONCEPTS

with the space environment would be necessary. The purpose of such analysis would be to evaluate the potential near-term and long-term impacts to human health and the natural environment. Evaluation of impacts to the sun has not as yet been considered, although disposal of waste by injection into the sun has been eliminated from consideration due to energy requirements.

Chemical Resynthesis^(a)

Draft pp. 1.1, 1.23-24, 3.2.1-23

Issue

Several letters noted that chemical resynthesis is not a disposal alternative but a treatment option. (154, 181, 208-NRC)

Response

In the final Statement discussion of the chemical resynthesis concept is presented in the chapter dealing with predisposal systems (4.0), specifically in the section on waste treatment alternatives (4.3).

Draft p. 1.23

Issue

Several commenters questioned why 600 m is selected as the disposal depth. (62, 218-DOI)

Response

The use of 600 m is an arbitrary depth selected from work done at the Carlsbad, New Mexico site. It is based on the depth (700 m) and the thickness (600 m) of the Salado formation. See Section 8.1 of the final Statement.

Draft p. 1.23-24, 3.2.1-23

Issue

One commenter noted that the potential processing problems associated with Chemical Resynthesis should be pointed out. (154)

(a) This concept was placed in the section discussing waste treatment and packaging alternatives in the final Statement (see Section 4.3.2).

ALTERNATIVE DISPOSAL CONCEPTSResponse

The fact that chemical resynthesis is a proposed waste form preparation technology that has undergone very little development, and that it would add processing complexity and produce secondary waste streams are discussed in the final Statement, Section 4.3.

Draft p. 3.2.2Issue

It is suggested that monazite may be a poor example to use to defend the mineralogic options, since monazite in nature is normally discordant, typically through the loss of uranium. (43)

Response

The extreme age of the monazites described (10^9 yrs) demonstrates that the fractional loss of uranium that may take place occurs at an extremely slow rate. This demonstrated stability is what makes monazite behavior relevant in the context of waste immobilization.

Draft p. 3.2.2Issue

The word pegmatite should be replaced by migmatite. The reference (Leonardos, 1974) specifically states that pegmatite contribution to monazite deposits are trivial (p. 1126). On p. 1127, Leonardos states "migmatites have supplied the material for the bands within the quartzite". Thus, migmatite is the term required to support the reference." (113-EPA)

Response

The correct terminology is incorporated into the Glossary of the final Statement (Chapter 8.0).

Draft p. 3.2.3Issue

This paragraph (second) does not consider the effect of radiation damage in the glass. This should be mentioned. (113-EPA)

Response

Information on the effects of radiation on glass can be found in the references given in Section 4.3.2 of the final Statement. The effects of radiation on glass in the presence of heat and water have been studied extensively and have generally been found to be minor.

ALTERNATIVE DISPOSAL CONCEPTS

Glass is a very radiation resistant material. However, a final decision regarding the use of glass as an acceptable waste form for use in a mined repository has not been made. Before the decision can be made, consideration must be given to many factors including the synergistic effects of radiation, heat, host rock chemistry, the presence of water, and the chemistry of the glass waste form. Current ongoing studies are addressing these and other important factors relative to the use of glass as the waste form component of the multi-barrier waste package.

Draft p. 3.2.13Issue

We are not interested in the integrity of the mineral (referring to detrital metamict grains), but rather whether or not the radioactive elements are retained within the structure. Of the minerals tabulated on Table 3.2.11, most, if not all, when analyzed by geochronologic methods are commonly discordant. (43)

Response

It is acknowledged that the mineral forms are discordant. However, the forms are attractive for incorporation of waste because their mineral integrity is considered to be an important attribute. The draft Statement pointed out in Section 3.2.3.2 that there is virtually no data base on the stability and insolubility of synthetic materials emplaced in the appropriate repository as far as the man-made elements are concerned. It would be necessary to assemble such information to provide confidence that the synthetic material waste form would comply with acceptable release rates such as those proposed by the NRC in its draft 10 CFR 60.

Draft p. 3.2.16Issue

The first two paragraphs are highly biased in favor of the synthetic mineral (as are the sources cited) and should be rewritten in a more objective manner. (58)

Response

The information on synthetic minerals has been revised for the final Statement (see Section 4.3.2).

ALTERNATIVE DISPOSAL CONCEPTSDraft Appendix PIssue

Appendix P represents, on the whole, original and innovative work on the promising concept of geologic emplacement following chemical resynthesis. However, the USGS has found numerous misspellings, incorrect formulas, etc. and suggest setting up a consultant group from universities and the USGS to review the concept and go over the CWMS sections on this approach thoroughly for accuracy and completeness. (218-DOI)

Response

The DOE recognizes the potential of chemical resynthesis and similar concepts that utilize analogs of naturally occurring minerals. It is supporting research on the concepts at DOE laboratories, several universities and the USGS. Their research will provide the accuracy and completeness the commenter seeks. The description in the draft Statement is only an overview of a concept that is still in the very early stages of development.

Very Deep HoleDraft pp. 1.24, 3.1.33, 3.1.34, 3.3.1Issue

Several commenters noted that in the summary section of the deep hole concept and in the discussion of the deep hole and mined repository concepts, the question of an appropriate disposal depth is continually alluded to but never directly addressed. (9, 36, 40, 121)

One commenter also expressed concern that too many holes would be required for the deep hole concept. (88, 121)

Response

Discussion of "How deep is deep enough?" is provided in Section 6.1.1.1 of the final Statement. Section 6.1.1.2 of the final Statement discusses the number of holes which would be required. The environmental impact of constructing the very deep hole is presented in final Section 6.1.1.6 and a comparison of this very deep hole concept with the other alternatives is presented in Section 6.2.

Draft pp. 1.24, and 3.3.1Issue

The question "How deep is deep enough?" (referring to the very deep hole and rock melt concepts) presents many unanswered questions. No geologist, geophysist or hydrologist can

ALTERNATIVE DISPOSAL CONCEPTS

say for certain how deep fractures and joint patterns extend that do contain some water, not only in sedimentary rock but also crystalline rocks. Even very small amounts of water could be vaporized by radioactive decay. The steam would travel upward, lose temperature until it became a liquid and then could contaminate the surrounding groundwater. This water in turn could contaminate a major aquifer within a large area. (9, 36)

Response

Revised and updated information regarding the question of "How deep is deep enough?" is presented in the final Statement. The question of migration of nuclides to the biosphere, via steam particles or other means, would be the subject of possible future technical studies for the very deep hole concept. It is important, however, to recognize that the pressure at depth is likely to prevent the formation of a phase fluid.

Draft p. 1.24

Issue

The last paragraph implies that rock structures can be determined to be unfractured. Is this implication accurate, or should the statement be qualified or rewarded? (124)

Response

The draft Statement acknowledges that rock characteristics, including fracturing, are not well known at these depths. The sentence in question was intended to suggest that a good site selection for a deep hole would contain predominant unfractured rock. Whether such a deep rock section can be located remains to be determined. See Section 5.1.1 and Appendix B of the final Statement for a more complete discussion of the uncertainties of geology and hydrology of depth.

Draft p. 1.25

Issue

Experience of rock bursts in deep mines (approaching 4 km depth) is cited on p. 1.12 in the geologic disposal section for mines 600 m deep but not here for deep holes 4 to 10 km deep. (181)

Response

The deep hole concept is somewhat different than the conventional geologic disposal concept in that the waste canisters will be placed directly into the drilled shafts. As a result, it is not anticipated that anyone will be physically inside these shafts at the depths considered for deep hole disposal. In addition these holes will have a drilling fluid in them to help keep the side wall material in place.

ALTERNATIVE DISPOSAL CONCEPTSDraft p. 1.25Issue

There are major problems with this option, if the hole is lost (collapses or is otherwise rendered unuseable) during the waste emplacement or backfilling-sealed stages, we would end up with the waste in the wrong place or irretrievably placed in an unsealed hole, both of which are probably unacceptable. These potential problems should be pointed out. (218-DOI)

Response

The problems mentioned in this comment are real and might be handled by casing the hole, as discussed in Section 6.1.1.2 of this revised Statement. Other potential problem areas are also discussed in the Section.

Draft p. 3.3.1Issue

It is stated in the draft: "In summary, the deep hole concept cannot be evaluated as a nuclear waste alternative without more information on the deep groundwater system, rock strength under increased temperatures and pressure due to decay of wastes, and the sealing of the holes over long periods of time."

These are three areas that have also been identified under the research and development needs (Section 3.1.6) for Conventional Geologic Disposal.

Why does the evaluation of deep hole disposal as an alternative depend on obtaining this information, while it is taken for granted that conventional Geologic Disposal is a viable alternative? If this information is obtained for conventional geologic disposal, would it apply to deep hole disposal? (208-NRC)

Response

The discussion from p. 3.3.1 referenced in this comment has been substantially revised and updated in the final Statement. In the final version (Section 6.1.1.1) the geologic characteristics of the site are presented in terms of the very deep hole concept, with the objective of locating the wastes below circulating groundwaters. In addition, the status of information available on very deep geology is discussed in Section 6.1.1.3.

In response to this question, it is worth noting the main difference between deep hole and mined repository disposal: deep hole disposal conceptually applies to depths which may not be practical for mined disposal and, hence, for geologic and hydrologic regions that might be quite different.

ALTERNATIVE DISPOSAL CONCEPTS

See a prior response in this section (General comment-Letter 208) for discussion of current status of knowledge of alternative concepts and the mined repository concept.

Draft p. 3.3.7Issue

Although fractures may be only a few meters long, they are often interconnected with others making a continuous flow network; therefore, the statement in paragraph 2, lines 7 and 8 in the report is misleading. (218-DOI)

Response

It is agreed that fractures are often interconnected; measuring/assessing this interconnection is an important problem of hydrology in fractured media and is the subject of several ongoing studies. The size and degree of interconnection of fractures is quite variable and may account for as much as 12 orders of magnitude variation in measured hydraulic conductivities.

Draft p. 3.3.7Issue

Fracture porosity should not be discounted. Fracture traces are systematically used in the exploration of oil and gas to at least 3 km. In areas of the crystalline shield, water well drilling commonly uses the concept of fracture traces to develop high capacity water wells. (43)

Response

DOE is in agreement with the comment that fracture porosity is very important, although to some extent the reviewer might be misinterpreting fracture porosity to mean fracture permeability. The latter was discussed in the subsequent paragraph on page 3.3.7 of the draft Statement. Deep resistivity measurements reflect only on interconnected porosity, which is quite low compared to the bulk porosity of crystalline rocks as measured in the laboratory. The difference in the two porosities is quite important to waste disposal; if water in the rocks is flowing so slowly that diffusion is important, then the entire interconnected porosity will provide large surface area for sorption/reaction. If the water is flowing rapidly, then only the small volumes of the fractures are of first-order importance.

ALTERNATIVE DISPOSAL CONCEPTSDraft p. 3.3.7Issue

In terms of fluid and solute migrate, fracture porosity might be the most important in many host rocks including basalt, granite and shale. Although there may be very few fractures, their permeability can be several orders of magnitude greater than that of the pores. (218-DOI)

Response

The statement in the draft Statement that fractures are seldom continuous for more than a few meters is misleading, except in the context of the paragraph, where the previous sentence states that ". . . permeability of typical geologic materials may vary over 12 orders of magnitude."

A detailed discussion of geology and hydrology has not been included in the final Statement. However, the final Statement does make reference to various sources of pertinent information in Appendix M.

Draft p. 3.3.7, Last Paragraph, lines 6-7Issue

This sentence appears misleading or erroneous; oil companies have tested many wells below depths of 500 m for permeability. If the reference is only to crystalline rocks, that should be made clear. (218-DOI)

Response

The comment is correct; the discussion is about crystalline rocks only, as noted on the caption to draft Figure 3.3.3 referenced earlier in that paragraph. This point is stated clearly in the final Statement. (Section 6.1.1.1.)

Draft p. 3.3.8Issue

Permeability measurements for one well in a sparsely fractured medium have little transfer value to the surrounding bulk medium. Measurements on many wells drilled at different angles are needed (which might compromise the site) or some new nonpenetrating method is needed (not yet developed). (218-DOI)

ALTERNATIVE DISPOSAL CONCEPTSResponse

The measurement of fracture permeability is recognized as a difficult problem and additional research will be needed to resolve remaining problems in its experimental determination for situations where it is an important factor. It is recognized from experience and has been experimentally shown that increasing compressive force or stress in a rock mass will decrease its permeability, including that contributed by fractures. At depths of 10,000 m the forces on the rock are substantial and the total permeability is expected to be extremely low. In view of lack of experience, experimental evaluation of this fact will be required before a final conclusion can be drawn.

Draft 3.3.11-13Issue

It appears that in the context of the duration of periods under consideration in repository planning, the discussion of rock strength needs to include at least in a general way changes in rock strength and characteristics that may occur with increasing time and their effects on permeability. (218-DOI)

Response

Section 6.1.1.4 (Issues and R&D Requirements) of the final Statement contains a discussion of thermomechanical and thermochemical factors relevant to the deep hole disposal concept. It notes that one of the areas where additional study is needed is in permeability changes caused by a rock mass being heated.

Draft p. 3.3.13Issue

The possibility that oxygen introduced with the waste will change the reducing conditions should be considered. (113-EPA)

Response

A discussion of the effects of waste heat and chemistry on the host rock is presented in the final Statement, Section 6.1.1.3, Status of Technical Development and R&D Needs, under the sub-heading Heat Transfer (Thermomechanical and Thermochemical Factors).

ALTERNATIVE DISPOSAL CONCEPTSDraft p. 3.3.30Issue

The fifth paragraph discusses the emplacement of a long column fuel canisters but fails to mention how crushing of the lower containers is prevented. The potential for crushing and the results thereof during the period of emplacement should be treated. (58)

Response

The final Statement includes, in Section 6.1.1.2, a discussion of measures required to prevent crushing of the waste package.

Draft p. 3.3.31Issue

In the third paragraph it is indicated that leakage considerations from the deep hole repository are similar to those for a conventional geologic repository. This comparison fails to consider that the transport shaft proceeds directly to point of emplacement, that it must remain open during the entire emplacement period and that the geometry of the repository would strongly affect thermohydraulic considerations. (58)

Response

There is no conceptual problem in a shaft proceeding directly to the point of emplacement if the seal, including the interface to the rock, has an integrity at least as good as that of the undisturbed host rock and if the seal were a multibarrier system of seals. A discussion of R&D requirements, including those necessary to develop borehole seals, for the very deep hole concept is presented in Section 6.1.1.3.

Draft p. 3.3.33Issue

If the permeability is less than as microdarcy, is not this permeability satisfactorily low? (35)

Response

If no very short interval in the hole had a greater permeability, if there were reasonable porosity, and if there were a low hydraulic potential gradient throughout the region of interest, then the answer would most likely be yes. One fracture in an otherwise "tight" rock mass can make considerable difference, thus the permeability cannot be an average over much hole length. And since permeability is only one component to the flow the flow equation, the other factors in the equation (hydraulic potential and porosity) must be taken

ALTERNATIVE DISPOSAL CONCEPTS

into account at any given site. However, conceptual and mathematical modeling will be required to analyze the impact of permeability on the consequences of operation. This type of analysis is important in defining the values of parameters that are satisfactory.

Draft p. 3.3.33Issue

Some discussion of retrievability from deep holes should be provided. (208-NRC)

Response

This issue is one that would require careful consideration before any waste could be emplaced using the very deep hole concept. A thorough analysis of failure modes and possible corrective actions, including retrieval or recovery of the waste, would be necessary. Methods employing overcovering or other means for retrieval would be analyzed. Although overcovering or other methods might be feasible for use, as currently envisioned, waste emplaced in a hole would not be retrievable.

Draft p. 3.3.33Issue

It is stated that, "It will be necessary to locate sites in strong, unfractured rock of low water content." This will exclude such media as shale and salt because of strength, and most other media because of fracturing. Why hasn't this same site selection criterion been applied to conventional geologic disposal? (208-NRC)

Response

In defining the very deep hole concept for the Statement it was necessary to make assumptions regarding the criteria for siting of a concept repository. The requirement for strong unfractured rock of low permeability was one such assumption. On the basis of current information about the characteristics of deep geologic systems, other assumptions might be equally valid. For example, emplacement of waste in very deep holes that terminate in salt, shale, or other media may be feasible. Research would be required however, to resolve geologic requirements for this concept. For the mined repository, however, many of the site selection criteria have been established on the basis of the large body of information available.

ALTERNATIVE DISPOSAL CONCEPTSDraft p. 3.3.33Issue

The section on thermomechanical behavior of rocks does not acknowledge that a significant body of information has been published on studies of hydrothermal alteration of natural rock bodies. The time, temperature, and the nature of ion migration in hydrothermally altered rocks has been studied for years by igneous/metamorphic petrographers, geochemists and mining companies. (208-NRC)

Response

The DOE acknowledges the existence of the significant body of information referenced by the commenter. However, this information pertains to much shallower depths than those envisioned for a very deep hole. The question of depth and its concomitant increasing pressure is the principal reason that significant information on the thermomechanical behavior of rocks is not available to support the concept. The information concerning hydrothermal alteration mentioned in the comment is available for rocks at depths where the total impact of the altering forces are less than that expected in very deep holes.

Draft p. 3.3.34Issue

Note \$500,000,000 just to drill the first hole; \$760,000,000 for R&D. Simple back-of-envelope cost estimates surely would show the costs of this system, even if it were possible to work out all the problems, to be prohibitive. (154)

Response

The cost of implementing a disposal option is a significant point of concern. Cost estimates that are accurate are dependent on the availability of thorough conceptual designs and a detailed understanding of the technical developments that must be completed. Although deep hole is an interesting concept and has a number of attractive advantages, detailed design studies or a thorough evaluation of technology requirements has not been undertaken. The comment that the cost may be unacceptably high may be correct. It will not be possible, however, to support this conclusion until these studies are completed.

Draft p. 3.3.36Issues

Is not borehole sealing covered under the shallow geological repository program? (35).

It should be reemphasized here that satisfactory backfilling-sealing techniques have not yet been developed and proven. (218-DOI)

ALTERNATIVE DISPOSAL CONCEPTSResponse

Section 6.1.1.3 of the final Statement discusses the R&D requirement associated with borehole sealing for the very deep hole concept. For very deep hole disposal, borehole sealing is potentially a greater R&D problem than for the mined repository because of the depth, pressures, temperature, and possible need for down-hole seal emplacement.

Draft p. 3.3.37

Issue

The citation for Reference 27 is inadequate. Provide information whereby Mr./Ms. Stevens can be contacted. (208-NRC)

Response

Victor Stevens is a mining consultant located in Salt Lake City. His address is 808 Kearns Building, Salt Lake City, Utah, 84101.

Issue

It would seem advisable, if not already considered, to gather information regarding the long-term stability of boreholes, wells, and other deep rock penetrations in regions considered favorable for repository location. These observations can provide additional clues on assessing the stability of the repository location. This would be useful in assessing the host media, as well as that of the overlying and underlying formations especially when considering the very deep hole concept of waste isolation. Perturbations of the earth's near-surface are readily detectable in both cased and uncased holes through sheared, ruptured, and squeezed boreholes and casings. (208-NRC)

Response

Gathering information on the long-term stability of existing holes has been considered. The final Section 6.1.1.3 contains discussion of geophysical logging and instrumentation techniques associated with data gathering of this nature.

Rock Melt

Draft p. 1.25

Issue

The introductory writeup on the rock melting concept does not present the disadvantages for this alternative, which were presented for the very deep hole concept, sub-sealed geologic disposal, etc. Equal treatment of all alternatives should be demonstrated in the final EIS. (208-NRC)

ALTERNATIVE DISPOSAL CONCEPTSResponse

The final Statement has attempted to develop an equal assessment treatment under a standard format to all alternatives, consistent with the state of knowledge available. The information in Section 6.1 is a factual presentation and Section 6.2 presents a comparison of the alternatives. In all cases an explicit comparison of advantages and disadvantages is presented.

Draft pp. 3.4.1-22

Issue

Rock Melting Concept--Would not radioactive heat act as a "pressure cooker" and cause an eruption of radioactive material into the environment? (88, 121)

Response

Sufficient rock is removed during construction of cavity that expansion of heated/melted rock can be accommodated. The point is a valid issue that must be considered. The understanding of the role that water might play in the melting of the rock and its availability to create high pressure steam is not well understood. For rock melting to be seriously considered as a disposal method, substantial R&D would be required to understand this phenomena and its associate operational difficulties. Some engineering features could be built into the rock melt repository to reduce the potential impact of such an event. The cavity itself can be excavated to a size such that thermal expansion of the rock can be accommodated, filtered vents can alleviate pressure buildup, cooling waste can assist in controlling temperatures of the mix, and waste can be added at a slow rate.

Draft pp. 3.4.1-22

Issue

A shortcoming of the description of the rock melt alternative is that no mention is made of the need for or availability of the water that's necessary for this alternative. Provide an estimate and discussion of the water requirements. (208-NRC)

Response

If the rock melt repository is co-located with a fuel reprocessing plant, little or no additional water (over that required for reprocessing) should be needed. Actual water requirements have not been accurately estimated; clearly, water availability will have to be one of the factors considered in the site selection process.

Section 6.1.2.2 of the final Statement indicates that the cooling water will likely be recirculated in a closed system where the steam driven off from the waste would be condensed and recirculated to cool the charge in the cavity. The closed cooling system would prevent

ALTERNATIVE DISPOSAL CONCEPTS

the release of radioactivity to the environment and minimize the amount of water required compared to a once-through system.

Draft p. 3.4.1-22

Issue

The Rock Melt Concept discussed in Section 3.4 assumes that the cavity is loaded over a period of years. This prolonged loading time has at least two disadvantages. First, the physical integrity of access and venting shafts must be maintained for the duration of the loading. Second, the cooling water itself will be contaminated and must be carefully contained and eventually the contamination must be disposed of as yet another waste.

Another loading scheme should be considered. The waste could be stored at the surface until the full load for the cavity has been accumulated. The waste could then be rapidly loaded into the cavity and the cavity quickly sealed.

It appears that the quick loading of the cavity is a practical alternative to the prolonged loading suggested in the GEIS. Further variations should also be considered, such as the use of an array of cavities (a few to maybe 10's of cavities). This would reduce the loading rate (in the case of the quick load) and distribute the heat load over a large volume. (208-NRC)

Response

While rapid loading of the cavity in the rock melt alternative has the advantage of minimizing the time that the access shaft must be kept open, it has the disadvantage of storing waste above ground for a long period of time. The risk of possible population exposure to radiation is considered to be higher from waste stored above the ground than in the case for waste downhole. The storage tanks required for approximately 40,000 MTHM of liquid high-level waste (33 million liters) which is necessary to fill each cavity (see Section 6.1.2.2 of the final Statement) is a primary consideration in the decision. Since the rock melt process was developed to handle a liquid waste steam the problem of disposal of contamination liquid (either residual from the waste stream or from evolving water) is one of the important factors which would eventually be addressed in the detailed engineering design studies of the process. See Section 3.4.2.1 and 3.4.2.2 of the draft Statement. In addition, further discussion is contained in Section 6.1.2.2 of the final Statement.

In addition, the problem of maintaining physical integrity of the cavity, shafts, and vents was identified as a disadvantage in Section 3.4.1.4 of the draft Statement and is similarly addressed in Section 6.1.2. of the final Statement. The approach of rapidly loading the cavity is not considered to be technically conservative or to be consistent with a step-wise approach, two requirements which are inherent to the DOE philosophy.

ALTERNATIVE DISPOSAL CONCEPTS

As presented in the final Statement (Section 6.1.2), the reference case use of 3 cavities, each approximately 20 m in diameter (6000 m³ volume) was selected on the basis of engineering judgment and would be subject to extensive review if the option were pursued. Factors which entered in the judgment were numerous; e.g. ease of excavation, desired spacing to prevent interaction between cavities, the likelihood of a geologic formation of appropriate size, etc. Many of the judgments were influenced by technical conservatism.

It should be noted that there is probably some as yet undetermined minimum size of cavity (corresponding to a maximum number of cavities) that will be required to accumulate a sufficient quantity of liquid waste. Smaller quantities, with very low attendant heat contents would not adequately melt the rock.

Further details of the basis for selection of the cavity size can be obtained from the various documents listed in Appendix M of Volume 2 of the final Statement.

Draft pp. 3.4.1-22

Issue

The treatment of "Rock Melt" in the GEIS misleads the reader as to the depth of investigation which has been completed. For example in the first paragraph on p. 3.4.4 of the GEIS, it is stated: "The concept has been assessed and reviewed (4,5) and preliminary laboratory scale investigations have been performed (6,7)." The workshop referred to as Reference 5, as productive as it may have been, fell far short of assessing "Rock Melt." The laboratory scale investigations were designed to study the descent of solid containers by rock melting, not the molten cavity concept. (208-NRC)

Response

Section 6.1.2.3 of the final Statement more properly describes the status of knowledge regarding the rock melt concept. It is agreed that the original referenced text was misleading.

Draft p. 3.4.1-22

Issue

While the potential for reduced cost certainly exists with this concept, it is hard to see how it could conceivably stand up to a "Circular 779" litany "uncertainties." In particular, we cannot conceive of NRC licensing this concept for spent fuel--criticality questions alone would doom the project. This entire section seems to be a rather blase treatment that not only reduces the concept's credibility but jeopardizes the credibility of the analyses of all the concepts. Some of the ideas and objectives are not bad, but achieving them with assurance seems very doubtful. This concept obviously lacks multiple barriers, control, retrievability, and opportunities for implementing contingency plans were things

ALTERNATIVE DISPOSAL CONCEPTS

to go wrong. We question the statement on p. 3.3.4 that "studies have identified no major technical issues which would cast doubt on the feasibility of the concept." This statement made elsewhere for other concepts may be plausible, but not for this one. Its expression here reduces its credibility elsewhere. (154)

Response

A fairly substantial re-evaluation was accomplished in the preparation of the final Statement. The comparison of concepts in Section 6.2 casts severe doubt on the credibility of rock melt based on current DOE waste management objectives and the status of knowledge regarding the concept. The concept would require resolution of significant questions in a number of areas (including criticality question) before it can be considered to be an acceptable candidate. See draft Section 3.4.1.4, Disadvantages and 3.4.2, Technological Issues Regarding Resoulation for Rock Melting. Isolation or Disposal has been rewritten to reflect current state of knowledge more realistically. The studies detailing the feasibility of the concept are referenced in the final Statement.

Draft pp. 3.4.4, and 3.4.12

Issue

Probable ground-water migration and circulation patterns associated with the rock-melting alternative need further consideration and discussion, preferable in conjunction with effects of thermal crack. (218-DOI)

Response

The need for additional data regarding the role of groundwater in the rock melt process is discussed in Sections 3.4.2.5 and 3.4.2.7 of the draft Statement in conjunction with other topics needing further study. More detailed discussion is presented in Section 6.1.2.3 of the final Statement. Many such questions need to be further resolved before this concept could be field tested with radioactive waste.

Draft p. 3.4.5

Issue

It is stated that retrieval of waste following emplacement would be difficult. This is understood, and not adequately addressed. (208-NRC)

Response

The commenter raises a valid point. It should be pointed out that the rock melt process was to make retrieval of waste as difficult as possible, since in one regard (safeguards) retrievability has been perceived as being equivalent to vulnerability. The current

ALTERNATIVE DISPOSAL CONCEPTS

DOE philosophy of providing for retrievability of waste in the event of some unanticipated event or repository shortcoming cannot be readily met for the rock melt concept. For this reason, among others, the rock melt concept is not considered worthy of further development at this time. The reader is referred to Section 6.2 for a more complete discussion of the situation.

Draft p. 3.4.6Issue

It is stated that the consequences of seismic activity appear minimal with proper facility design. Discuss the effects of seismic activity on surface facilities supplying cooling water and cleaning up the steam, and on the reliable supply of cooling water to the waste. (208-NRC)

Response

A waste repository should be located against rigid siting criteria that are assumed to specify an aseismic area. Moreover, the surface facilities should be designed to survive significant (probably about 0.5 g) ground accelerations with complete containment of radioactivity. If water to the waste cavity is shut off, the facility should be designed so that the emplacement shaft will close automatically, and the rock melting process will start. Thus, the cavity would not be available for additional waste (early melting started); however, this is not serious economically and should have no significant effect on safety. Before rock melt could be seriously considered for a disposal operation the impact of seismic events would have to be considered. Empirical observations indicate, that unless an underground structure is transversed by a capable supporting surface equipment, such as cooling and steam clean up equipment, it would have to be designed to withstand a seismic event appropriate to the region in which the facility is located. This would be in consonance with the precept that waste disposal facilities would be required to meet appropriate standards which are applied to other nuclear fuel cycle facilities.

Draft p. 3.4.9Issue

A major point missed with rock melting is the consequent melt cooling. Differentiation will result, and the last formed liquids will concentrate elements such as uranium. This will form late hydrothermal liquids of extreme radioactivity. Whether or not this might result in criticality should be investigated. (43)

ALTERNATIVE DISPOSAL CONCEPTS

Response

This point is addressed in Section 3.4.2.8 of the draft Statement, Study of Criticality Potential in Rock Melting (p. 3.4.8). As pointed out in Section 6.1.2.2 of the final Statement, only the use of liquid high level waste from reprocessing is assumed for the reference case, and this waste is relatively low in fissile isotopes. Thus, in the reference case, spent fuel is not considered due to "uncertainties associated with emplacement, such as additional criticality concerns..." While it is not yet accepted unequivocally that the last formed liquids will concentrate uranium, this is a distinct possibility. Other possible mechanisms can be postulated also, so that the concern over criticality is quite pertinent.

Draft p. 3.4.10

Issue

Figure 3.4.4 does not present the temperature profiles that are necessary to completely characterize the extent and duration of the thermal load on the host media. The maximum increase in temperature at the earth's surface can occur hundreds of thousands of years later than shown. (Numerical models can be very costly to run for long times and distances require, however any analytic model is available. See Reference 3 of Appendix C of TID-28818 (Draft), "Subgroup Report on Alternative Technology Strategies for the Isolation of Nuclear Waste.") (208-NRC)

Response

Figure 3.4.4 was intended to be only illustrative of the relatively early temperature distribution around a rock melt repository. Future engineering studies would investigate this topic in the necessary detail. In addition, site-specific calculations should be made to determine the effects of different strata, aquifers, etc.

Draft p. 3.4.13

Issue

If operations as unattractive as grinding solidified HLW to a powder are proposed, the full impact of those operations should be included. (58)

Response

This commenter raises an important point regarding the rock melt information in the draft Statement. The potential approach questioned has been eliminated from the reference concept in the final Statement since there is no obvious reason to first convert liquid high-level waste to glass and then to convert that to a powder. The high-level liquid itself has the desired properties for direct emplacement in this concept.

ALTERNATIVE DISPOSAL CONCEPTSDraft p. 3.4.13Issue

Handling equipment for these operations will be formidable. (35)

Response

The point made in this comment (regarding Section 3.4.4.2 of the draft Statement) accurately indicates potential problems with the proposed handling method. At this point, the method is only a proposal and has a high probability of being rejected on a more thorough engineering review and analysis. In fact, the final Statement does not incorporate these approaches into the reference system. If the rock melt disposal technique is to be used, there would be no obvious reason to convert to glass before emplacement, and the as-produced liquid high-level waste would be emplaced directly. Similarly, there appears to be no current interest in emplacing solidified canisters of HLW in the rock melt cavity.

Draft p. 3.4.15Issue

Extended subsurface storage of waste would be very difficult to implement. What would it accomplish? (35)

Response

Extended subsurface storage of waste was discussed to identify the potential options available in the operation of the disposal method. There is no particular known advantage to this approach, and it is believed that other methods are much more satisfactory. In the final Statement (Section 6.1.2) it is pointed out that retrieval of the HLW after emplacement is considered very likely to be difficult or impossible. This is a primary reason for rating the potential of the rock melt cavity to be very low, as measured against DOE program objectives.

Draft p. 3.4.17Issue

The post sealing period environmental effects are assumed to be "the same for (non-salt) conventional and Rock Melt repositories." The basis for this assumption should be given. If the thermal barrier effect protects the HLW from groundwater leaching for possibly a few thousand years, might not the post sealing performance be superior to that for conventional geologic disposal? (208-NRC)

ALTERNATIVE DISPOSAL CONCEPTSResponse

The assumption that post-sealing environmental effects would be the same for the rock melt process as for the mined repository was re-assessed subsequent to the issue of the draft Statement and the conclusions are presented in final Sections 6.1.2.5 and 6.2. The current DOE position is that there are substantially more uncertainties relating to a performance assessment capability for rock melt and that the expected performance is less than for a mined repository, especially during the period when the waste-rock mixture is molten. Subsequent to solidification, the environmental effects are dependent on factors such as the immobility of the resultant waste forms and the hydrologic regimes in existence at the time. Because there is a lack of multiple barriers, the concept is not consistent with DOE policy. More refined assessments will be made as the rock melting concept is developed. Many potential environmental effects will be sensitive to individual site parameters.

Draft p. 3.4.18Issue

The reasons for post-operational monitoring of melt growth, etc. for prolonged periods should be explained along with any response that might be made to the information gained.
(58)

Response

Post-operational measurements will be carried out during R&D phase to validate calculational results obtained from modeling. Post operational monitoring following an actual full-scale application would be an activity that would maintain confidence and also increase knowledge of the disposal operation. The information would be useful in building similar facilities and, in case of unplanned events, to provide guidance to the nature and magnitude of the problem. The technically conservative approach suggests that post emplacement monitoring should be desirable in any disposal option until such time as it is operating satisfactorily and the lack of any unanticipated events is verified.

Issue

Information on the possible failure of the cooling system, mitigative actions, and environmental impacts should be provided. (208-NRC)

Response

Information that is available has been presented in Section 6.1.2.2, 6.1.2.3, and 6.1.2.4 of the final Statement. Because the sections are relatively lengthy, no detailed answers are provided here, but the commenter is referred to those sections.

ALTERNATIVE DISPOSAL CONCEPTSIssue

In the event that the cooling system for the waste fails while still needed, it will be very difficult to repair because of its proximity to the waste. (208-NRC)

Response

The commenter addresses a valid concern. DOE's approach would be the incorporation of independent cooling systems to the design. This is discussed in Section 6.1.2.2 of the final Statement. Another would be to design a "fail-safe" mode of operation where the emplacement shaft will automatically close upon stoppage of cooling water so that rock melting would commence at that point and the cavity would be sealed. Either approach is considered technically conservative.

Island DisposalDraft p. 1.26Issue

The statement on line 9, that island arcs are highly active seismically and volcanically is not necessarily correct as there are tectonically inactive island arcs. (208-NRC)

Response

The text in the final Statement (Section 6.1.3) has been revised to note that island arcs, at ocean continent margins, are frequently active seismically and volcanically.

Draft p. 1.26Issue

Other unresolved problems or disadvantages that should be mentioned regarding island disposal are: greater probability of disruptive geologic events (faulting, earthquakes, volcanism, etc.); perhaps greater erosion rates; more subject to higher rainfall rates, tropical storms, and tsunamis. (218-DOI)

Response

In the preparation of the final Statement efforts were made to provide a balanced discussion of the advantages and disadvantages of the concept using the most current information available in the technical literature. Accordingly, for relatively undeveloped concepts, many of the issues that would be the subject of future R&D are not addressed. An example of such an issue is rainfall rates that would be expected to be site specific although very important to the evaluation of island hydrologies. No island sites have been considered by the DOE. Therefore, rainfall rate data are not available.

ALTERNATIVE DISPOSAL CONCEPTSDraft Section 1.3.5Issue

Section 1.3.5 states that, "Salt deposits are unlikely to be available at island sites; the most probable disposal formation (sic) is crystalline rock." From this statement one would conclude that crystalline rock was the most common rock type exposed on islands. This is not the case; e.g. the Antilles, the Japanese and Phillipine archipelagos, New Guinea, Bikini, Bermuda, etc. (208-NRC)

Response

The statement does not suggest that crystalline rock is the most common type exposed on islands; rather it is the most probable disposal formation.

Draft Sections 1.3.5 and 3.5Issue

In view of the statements:

"The DOE Task Force Rough Draft Report states that the DOE has no program to actively investigate the concept." (p. 3.5.1)

"Institutional and political aspects of acceptance of either U.S. or foreign island sites have not been addressed and are neither suggested nor implied." (p. 3.5.2)

it is not clear why this "alternative" is included. (154)

It is nowhere stated as to why this alternative is being considered. (34)

Response

Island disposal is a potential disposal method which has been suggested as a candidate for consideration in the alternative waste program. This disposal concept is a variation of the mined repository disposal concept in that the waste would be entombed in a deep geologic formation and is therefore a valid concept for consideration. Section 6.1.3.1 in the final Statement briefly presents a discussion of the attributes and disadvantages of this disposal option. A comparative assessment of the eight alternatives and the mined repository is presented in Section 6.2 of the final Statement.

ALTERNATIVE DISPOSAL CONCEPTSDraft Section 3.5Issue

This concept in a way is not an alternative to geologic disposal but simply geologic disposal in a specific place. (154)

Response

It is recognized that island repositories would be mined repositories sited at island locations. However, the unique hydrologic and geologic settings of islands and the ocean transportation requirements are a sufficient departure from the continentally located mined repositories to warrant separate consideration of the concept. The differences between the two concepts are clarified in Section 6.1.3.3, System and Facilities Description, and in Section 6.1.3.3, Technical Issues, of the final Statement.

Draft Section 3.5Issue

Island Disposal--Too costly, sea transportation, security system, how will the island be protected by our military men? What will be the cost to the taxpayers? (88, 121)

Response

These comments are addressed in the comparative analysis in Section 6.2 of the final Statement. This statement also points out that although detailed cost estimates for construction, operation, and decommissioning have not been made, it is presently estimated that these costs would be at least double those for a continental mined repository (see Section 6.1.3.6 of final Statement).

Draft Section 3.5Issue

The discussion in Section 3.5 indicates that two options for Island disposal are being seriously considered. One option is disposal in oceanic islands for which relatively long sea voyages for transporting the radioactive wastes will be necessary. The other option is disposal in continental islands. For this option, the transport time at sea is small with the possibility of using a ferry-type transport system; facilities at the embarkation and receiving port could be simplified. The two options should continue to be treated separately and additional information concerning environmental impacts and accident risks be developed for both options. It is important to continue to explore both options with the ultimate choice being left to a risk benefit analysis after more complete information is developed. (208-NRC)

ALTERNATIVE DISPOSAL CONCEPTSResponse

The differences between oceanic and continental islands do not justify a totally separate treatment at the current stage of development for this concept.

Draft Section 3.5Issue

The ability to dewater a site is an extremely important site characteristic. Dewatering with the attendant equipment may impose such an economic burden that an other-wise suitable site may be ultimately rejected. The dewatering problem may, in the end, result in the rejection of the island arc and oceanic island locations. In addition, the retrievability of waste placed in any island watery environment, particularly salt water, is questionable considering the effects of corrosion on dewatering equipment. (208-NRC)

Response

One of the criteria for selection of a site would be for a host rock of low permeability. Thus, major dewatering problems during construction would be unlikely to occur at a site suitable as a repository location.

Draft Section 3.5.1Issue

While this option approaches in effectiveness some technological advantages of the conventional geologic option, it offers additionally an international approach that is attractive. However, detailed studies of this option have not been made, and it is now merely a concept lacking a strong consensus for adoption. (124)

Response

As noted in Section 6.1.3.1 of the final Statement the DOE does not currently have a program to actively investigate the concept. However, the DOE believes that information presented in the final Statement provides a sufficient basis for assessment of the attributes and disadvantages of alternatives and mined repository. Section 6.2 of the final Statement presents a comparative assessment of the disposal technologies.

Draft p. 3.5.1Issue

The "possible advantageous hydrogeological" features do not stand out in reading Section 3.5. (124)

ALTERNATIVE DISPOSAL CONCEPTS

Response

The "possible advantageous hydrogeological features" are discussed in Section 3.5.1.1 under Groundwater Transport: Freshwater Lens Location and Groundwater Transport: Saline Lens Location of the draft Statement. However, due to the lack of site-specific data, the Statement cannot go into detail concerning hydrogeological features. From a generic description point of view, the Statement does clearly depict the present understanding of the isolation features inherent with the island hydrogeology. These features may represent favorable isolation barriers. Section 6.2 of the final Statement discusses advantages and disadvantages of the alternative concepts, including the island disposal concept, and the mined repository.

Draft p. 3.5.1 and 3.5.5

Issue

The assumption of a "practically static" salt water system below the fresh water lens should be approached with reservation. The stability depends upon many factors some of which are mentioned in the text (p. 3.5.19), some aren't. Examples of these factors are: amount of rainfall, frequency of rainfall, water usage (pumping regimes), tides, sea level fluctuations, and erosion. In what sense is the ocean considered to provide an additional barrier? (208-NRC)

Response

The ocean is not considered a barrier. However, radionuclide leachate bearing groundwaters that might discharge into deep oceanic waters would be expected to be diluted and dispersed to very low concentrations. Shallow water releases would be expected to cause greater impacts than deep water releases. Conversely, while dispersion of the radionuclides in the ocean might occur, concentration could also take place in the food chains, especially in littoral and coastal areas.

Draft p. 3.5.6

Issue

It is not clear how the need for 6 to 10 island repositories was developed. (124)

Response

The final Statement draws no conclusions regarding differences in the required numbers of island repositories as compared to continental mined repositories. However, the final Statement does note that physical limitations on useable island areas might cause a need for a greater number of island repositories than Continental mined repositories. For a comparison of the alternatives and the mined repository see Section 6.2 of the final Statement.

ALTERNATIVE DISPOSAL CONCEPTSDraft p. 3.5.12Issue

The statement that 85 percent of the world's earthquake energy is released in the Pacific margins should be documented. (208-NRC)

Response

The sentence referred to in the comment was taken from Physics of the Earth's Interior, (Bott 1971). "About 75 percent of world-wide shallow earthquakes (focus 0 to 70 km depth) and 90 percent of a world-wide intermediate and deep earthquakes (focus greater than 70 km depth) occur beneath the circum-Pacific margin belt of island arcs, deep trenches and mountain ranges." The reference is included in the Reference List for Section 6.1 of the final Statement.

Draft p.3.5.12Issue

Figure 3.5.6 does not show major basement rock types. There is a figure showing major basement rock types in Reference 5 (Bayley and Muehlberger 1968), which has Figure 3.5.6 as an inset, titled "Principle Basement Provinces." (208-NRC)

Response

Figure 3.5.6 of the draft Statement should have been titled "Principle Basement Provinces Along the Pacific and Atlantic Coast."

Draft p. 3.5.16Issue

An unequivocal statement that island arcs are completely unsuitable for waste disposal because of seismic activity would be in order. (124)

Response

Some island arcs are seismically inactive. However, before any island were chosen for the disposal of waste thorough evaluation of the geology and hydrology would be required. Islands found to be in geologically active areas would be eliminated for the same reasons that tectonically active continental areas have been eliminated from consideration for siting of a mined repository. See Section 5.2.1.1 of the final Statement for a discussion of long term geologic stability.

ALTERNATIVE DISPOSAL CONCEPTSDraft p. 3.5.16Issue

The reference to measurements in deep artesian wells in paragraph 3 seems inconsistent with the concept of a floating freshwater lens, because conditions in a confined or artesian aquifer might not necessarily reflect the freshwater/saltwater ratio. Influences other than differences in density can be effective in confined aquifers. (218-DOI)

Response

The term "artesian" should be deleted from the referenced sentence. The commenter correctly notes that a hydrologic system characterized by a freshwater lens floating or deep saline water would not support artesian aquifers.

Draft p. 3.5.17Issue

Second paragraph: "Of lesser significance are local alluvial (and other sedimentary) deposits..."

This is true if "significance" refers to abundance. However, in terms of importance the sedimentary deposits are of great significance because, with the lava flows, they provide the dateable units for defining tectonic history and predicting future activity. (154)

Response:

The intent in the draft Statement was to discuss significance in reference to aquifers. It is acknowledged that alluvial (and other sedimentary) deposits would be of importance in characterizing the geologic history of the island.

Draft p. 3.5.18Issue

The discussion of sorptive phenomenon is not sufficiently covered. A comparison of the sorptive properties associated with island disposal with those associated with conventional geologic disposal should be presented, to determine if the multibarrier approach has been effectively utilized. (208-NRC)

Response

Data on the comparative sorption properties associated with the island disposal option compared to mined repository disposal are not presently available. However, some

ALTERNATIVE DISPOSAL CONCEPTS

information on the absorption properties of deep sea sediments is contained in Section 3.6.3.2 of the draft Statement of the subseabed geologic disposal alternative.

Draft p. 3.5.19Issue

Figures 3.5.9 and 3.5.10: "Isolation" in the Figure titles should be "Containment" (our terminology). (154)

Response

Please refer to the Glossary, Chapter 8.0 of the final Statement.

Draft p. 3.5.19Issue

It should be noted that the dispersion and diffusion may be very active in this type of system, especially in combination with a natural zone of dispersion along the saltwater/freshwater interface. (208-NRC)

Response

A study of the mechanisms of dispersion and diffusion for radioactive species within the geohydrologic setting of island repositories would be a necessary part of a development program for this concept. It is likely that emphasis would be placed on the freshwater/saltwater interface and the sorption and transport characteristics of the boundary region. The effects of a moving interface would also be considered.

Draft p. 3.5.20Issue

Two disadvantages of the island concept at the bottom of the page the following should be added:

- Because of ocean cover and limited stratigraphy, the tectonic activity will be difficult to document with adequate assurance of safety relative to faults and earthquakes.

Fifth paragraph: "The accuracy of geological information at actual data points (drill holes, surface exposure, underground workings) will be good. However, accuracy of the spatial extrapolation will depend upon the density of available data in space and time, the complexity of the geology, and the knowledge and experience of the scientist performing the work. Similar comments apply to oceanographic data with an additional proviso that many

ALTERNATIVE DISPOSAL CONCEPTS

oceanographic phenomena are constantly varying in time, thus requiring more intensive data collection." These statements are of such fundamental importance, that they should not wait for expression in this section. Rather, they should be expressed right up front in the Summary or Background sections together with other similar truisms of geology. (154)

Response

The commenter addresses uncertainties that are currently applicable to the island disposal concept. Section 6.2 of the final Statement discusses the influence of uncertainty on the selection of preferred alternatives for continued development and presents a comparative assessment of this concept.

Draft p. 3.5.20

Issue

The disadvantages of island sites should include the sociological objection to the use of continental islands and concern for volcanism on oceanic islands. At least in the U.S. context, continental islands are cherished by the environmentalists and proposals to use them as a nuclear "dumping ground" could be expected to raise a monumental furore. Similarly, although theory would suggest that some islands of volcanic origin will remain inactive for geologic time spans, the critics of nuclear power and the media would have a field day decrying the 'faulted logic' of the nuclear advocates. Information should be presented to show how islands of volcanic origin will be proven to be inactive. (58)

Response

Section 6.1.3.4 of the final Statement addresses the impacts to natural system that might be expected if the island repository concept were to be used. Section 6.2 presents a comparative assessment of alternative concepts and the mined repository. This assessment includes consideration of the differences in impacts between alternatives. The volcanic inactivity for oceanic islands would be based on geologic data on the time of the most recent eruption. As shown in Figure 3.5.7 of the final draft Statement, the age of the oceanic islands and time since the last eruption can be shown in many instances to increase with distance from a hot spot or mid-oceanic ridge.

Draft p. 3.5.21

Issue

Another disadvantage should be mentioned. Islands are often more associated with resources and more subject to faulting, seismicity, volcanism, erosion, effects of sea level changes, extreme storms, and tsunamis. (218-DOI)

ALTERNATIVE DISPOSAL CONCEPTS

Response

Such considerations are presented in Section 6.1.3.4 (Subpacts on Natural System Impacts and Resource Consumption) and Section 6.1.3.5 of the final Statement.

Draft p. 3.5.21

Issues

It is noted that the transportation link from the mainland to the island would involve an additional potential for accidents. It would seem appropriate to note that this increase is not necessarily linear with distance, in that the most common location for accidents would be near one or the other ports, and that ship accidents are strictly proportional to the number of port calls made. (124)

Response

The final Statement has been revised to reflect the point made in this comment. The potential for accidents will increase although not necessarily linearly as the distance to the island increases. Reference is made to Section 6.1.4 (subseabed concept) of the final Statement for additional discussion pertinent to sea transportation as it applies to concept in terms of requiring over water transportation of the waste. Section 4.5 of the final Statement presents a comprehensive discussion of transportation impacts.

Draft pp. 3.5.21, 23, 24

Issue

The hydraulic gradient and the stability of freshwater lens may also be affected by the sea level slope induced by coastal currents and topographic focussing of surface and internal gravity waves. (23-DOC)

Sea level slope is another factor in the stability of freshwater lens. (23-DOC)

Factors of importance to sediment patterns and movements include tides, land discharges, coastal circulation, littoral currents induced by surface gravity waves, and direct wave actions during storm and tsunami. (23-DOC)

Response

Stability of the freshwater lens may be affected by changes in sea level, glaciation, thermal convection effects, repository construction, changes in climate and precipitation, modifications of surface runoff, and by changes in the sea level slope induced by coastal currents and topographic focussing of surface and internal gravity waves and other effects.

ALTERNATIVE DISPOSAL CONCEPTS

Sediment patterns may be affected by changes in sea level, glaciation, changes in climate and precipitation, modification of surface runoff, tides, littoral currents induced by surface gravity waves, and direct wave actions.

Draft p. 3.5.22Issue

Report should state whether or not prior chemical or other disposal has ever been conducted on island sites and what the results were. (124)

Response

If selected as a viable option the island disposal concept would require in-depth R&D support of a thorough licensing process. Information regarding releases from such disposal sites, if they exist and if the geologic and hydrologic settings are sufficiently analogous, would be useful. However, there is little likelihood that information is in existence for waste deposited in selected hydrologic regimes deep within an island geology. No such information was used in the preparation of the final Statement.

Draft p. 3.5.23Issue

Under Section 3.5.2.2, some estimate should be provided of the probability of accidents on the sea lanes, which might lead to loss of the radioactive cargo. Cost estimates should also be provided. (208-NRC)

Response

Estimates of sea transportation accident probabilities were not undertaken in this study. Data is undoubtedly available and could be assembled and interpreted. Work done in Europe, especially by the U.K. and in Japan in relation to spent fuel sea transportation, may be available. In the U.S. the American National Standards Institute working group, ANSI N552, is currently considering standards for the water transportation of radioactive (spent nuclear reactor fuel) materials. These standards were not complete at the time of the final Statement.

Detailed costs have not been estimated. For further discussion of cost analysis see Section 6.1.3.6 of the final Statement.

ALTERNATIVE DISPOSAL CONCEPTSDraft p. 3.5.25Issue

Replace the sentence beginning with: "However," on line 13 with: However, their natures and time variant behaviors, particularly of subsurface current, are incompletely understood. (23-DOC)

Replace the sentence beginning with: "Similar" on line 15 with: For oceanographic data there is need for both intensive and extensive observations over a long period in order to resolve the time and space scales of the coastal, littoral, and other flows. (23-DOC)

Response

The above sentences are preferred substitutes for the applicable text of the draft Statement.

Draft p. 3.5.27Issue

It should be noted that current models are not able to accurately predict flow through fractured media, which will be normally encountered in islands of volcanic origin. (208-NRC)

Response

Models for accurately predicting the migration of radioactive species through fractured media are not currently available and basalts that are extruded to solidify under the ocean are normally highly fractured. Also, as noted in Section 3.5.1.2 of the draft Statement, islands of volcanic origin have highly fractured basalts that may be bounded by sheets of dense impermeable rocks (dikes). Detailed characterization of the host geology would be a necessary part of the development of the island concept. Until characterization of the host geology for an island was completed, the modeling needs for groundwater flow would not be known.

Draft p. 3.5.29Issue

Section 3.5.6.2 implies that there are no uncertainties associated with island disposal beyond those associated with conventional geologic disposal, an implication that appears simplistic. The areas of uncertainty should be summarized and some quantitative assessment of their potential consequences provided. (58)

ALTERNATIVE DISPOSAL CONCEPTSResponse

The primary area of uncertainty and difference between island disposal and continental geologic disposal is the geohydrology of the island. In addition, the tectonics of oceanic islands or island arcs would be different from continental locations. The rock mechanics, waste container, waste form aspects, etc., however, would be expected to be very similar to those for conventional mined disposal in crystalline rocks.

Draft p. 3.5.29Issue

Section 3.5.6.3 identifies research and development areas that need to be explored in order to resolve uncertainties in island disposal. One area is the level of risk associated with extended sea transportation paths. Since the complexity of port facilities varies with the island disposal option being considered, the level of risk, both in terms of routine occupational exposure and exposures due to accidents, should also be considered as an area needing development. (208-NRC)

Response

Section 6.1.3.4 in the final Statement includes consideration of port facilities. Also, reference is made to the subseabed alternative (Section 6.1.4 of the final Statement) for additional discussion of the current development status and R&D needs for transportation of waste. In the presentation of comparative analyses, Section 6.2 of the final Statement includes the transportation aspects of the various alternatives.

Draft p. 4.15Issue

It is not accurate to state that the insular geologic surroundings are of inherently dynamic nature. This is not so especially for the east coast continental islands. East coast islands are probably less likely to contain, or be near, valuable resources than some of the west coast islands, thus lessening the possibility of repository intrusion. (208-NRC)

Response

The referenced text from the draft Statement was included in draft Section 4.5.3.2, Susceptibility to Natural Phenomena. The dynamic nature of insular geologic surroundings refers to the natural processes of geologic and meteorologic changes. These changes are pronounced at continental boundaries where the erosional effects of the ocean may combine with tectonic changes to make island geologies and hydrologies inherently unstable.

ALTERNATIVE DISPOSAL CONCEPTSDraft 4.20Issue

Table 4.5.2 presents preliminary estimates of the socioeconomic impact of the waste management options. An assumption stated under island disposal is that dockside shipping facilities will be constructed in a well established port area. For the no recycle option, packaged spent fuel will be shipped to the island disposal area. The recent NRC interim rule for safeguarding spent fuel shipments may prevent the use of well established port areas so that the conclusion reached, that the incremental impact is small, may not be valid. (208-NRC)

Response

Rules and regulations regarding safeguards and security for the shipment of spent fuel and high-level waste are prescribed in the Code of Federal Regulations Title 10, Part 23. Paragraph 73.37 prescribes rules for routing of such shipments. These rules do not preclude properly secured shipments to existing port areas.

Subseabed DisposalDraft p. 1.26-29Issue

The section on sub-seabed disposal is the most complete and concise section in the summary. It describes a concept that appears highly favorable from a geologic viewpoint and concludes that "five years of research have revealed no technological reason why further development of the sub-seabed disposal option should be abandoned..." Considering that this concept is the second most favorable one, the conclusion should be stated positively rather than negatively. Also consideration should be given to simple disposal on the seabed, as repropose by Dr. Bernard Cohen. (154)

Response

The change in emphasis requested by the commenter has been brought into the final Statement (see Sections 1.4 and 1.8). Seabed emplacement techniques have been under study for several years. The example or reference method utilized in this Statement is a free-fall penetrometer. All alternatives to this technique (cited in this Statement) involve penetration or emplacement beneath the ocean sediment. Section 6.1.4.3 of the final Statement points out that penetrometers are widely used in Marine, land, space, and arctic operations.

ALTERNATIVE DISPOSAL CONCEPTSDraft p. 1.27Issue

If there is sufficient heat to modify the red clay of the ocean floor, there may be sufficient heat to initiate convection currents in the overlying water. If sufficiently large in a real extent, this would cause an upwelling, bringing to the surface material from the lower depth of the ocean and possibly from the ocean floor. This material could be nutrients, inert material, or if a canister ruptured on impact, radioactive material. (218-DOI)

Response

The red clays should not be modified if the temperatures are kept below 250°C. The organic carbon content of these clays is less than 0.1%. Thus, the nutrients which will be released will be minimal. Calculations indicate that convective currents in the sediments will move a water molecule only 3 m in 1000 years. Convection cells in the water column will not occur.

Draft Section 3.6Issue

More emphasis should be placed on understanding the nature of transport of materials in water column in areas under study for waste disposal. Ongoing programs in quantifying biological pathways should be expanded and comprehensive program of physical, chemical, and biological measurements should be undertaken, and models developed for deep ocean layers. It is essential to quantify what will happen in the case of accidental breakage or unexpected leakage in terms of the water column serving as an emergency barrier. (23-DOC)

Response

The Subseabed Disposal Program Plan (Sandia 1980a) details projected or ongoing studies regarding transport in the water column via physical oceanographic processes as well as biological processes. Also discussed there are the impact of radionuclides on biota as well as the potential impacts on man if radionuclides were to get into the water column or to the ocean surface.

ALTERNATIVE DISPOSAL CONCEPTS

Draft Section 3.6

Issue

Somewhere in this section several other matters should be briefly considered: How deep would the projectiles be sent? What distance beneath the sediment surface, and how far from the rock beneath? What about the concept of recovery, if unforeseen dangers are found to exist? (113-EPA)

Response

A methodology for canister emplacement is proposed to be determined during Phase 3 (Engineering Feasibility) of the Subseabed Program current system pursued by Sandia National Laboratory for the DOE. Questions regarding distances below the sediment surface and above the basement rocks are to be answered as part of Phase II (Technical and Environmental Feasibility) which is now in progress. Firm answers are not yet available. It appears at this point, however, that a minimum of 30 m of sediment may be needed to separate a canister from both the sediment surface and the basement rocks. Also see Emplacement and Retrieval/Recoverability in Section 6.1.4.2 of the final Statment.

Draft Section 3.6

Issue

This section is very interesting and very well done. The concept is probably the safest one with regard to geologic factors. Some basic truths about necessary elements for demonstrating tectonic stability are expressed here that not only apply to other concepts as well, but which should be expressly set forth as guidelines in the Summary or Background section of the Final EIS. For example:

p 3.6.2 ". . .ability to make long-term predictions of stability and uniformity on the basis of . . . sediments that have been accumulating for 70 million years."

p 3.6.3 "The more predictable and uniform the geologic environment, the less detailed the specific site studies must be to determine the properties of the geologic formation."

". . . areas where processes are slow and continuously depositional, and where tectonic processes have been and are predicted to be minimal for millions of years, are the most uniform and pedictable on the globe."

p 3.6.11 ". . . areas with the greatest uniformity and predictability will be those where uplifting will not occur again for millions of years and where deposition is continuous and uniform." (154)

ALTERNATIVE DISPOSAL CONCEPTS

While it is not well enough advanced to be considered for an initial repository, this concept appears to have enough promise that it should receive some R&D funding. (154)

Response

Although DOE would like to restrict ourselves to only the most stable, uniform, and predictable areas, adequate areas can probably be found wherever natural uplifting and reprocessing processes have not occurred. One must be very careful to delineate overall global processes and to show where and why the highest reliability can be found.

The subeabed program is currently being funded by the DOE. Fiscal year 1980 funding was for \$5.9 million.

Draft Section 3.6

Issue

The option has been under active development since 1973, and is considered one of the leading alternatives to the conventional geologic option. The development effort now includes four other countries, which recognize that an international approach may be necessary to an effective solution. (124)

Response

Six countries are now included in an NEA/OECD working group on subseabed disposal of radioactive waste: U.S., Netherlands, Japan, Great Britain, France, and Canada.

Draft Section 3.6.2.3

Issue

This section starts off with the identification of the barriers to the movement of radionuclides, then fails to discuss two of them: "any controlled modification of the medium," and "the benthic boundary layer." A discussion of these barriers should be provided. (208-NRC)

Response

The various barriers are described briefly in the Section 6.1.4.2 of the final Statement under Waste System Description.

ALTERNATIVE DISPOSAL CONCEPTSDraft Section 3.6.3.6Issue

Subseabed geologic disposal--Legal and political issues will be raised. Why should Japan and Russia allow us to dump our radioactive waste in that area? (88, 121)

Response

No country, under present international law, has any right to control the activities of another country upon the high seas. Japan and many other maritime nations, since there are no stable geologic formations suitable for waste disposal on the islands proper, will probably be required to safely dispose of their waste either in the international ocean regime or rely upon some other nation to dispose of Japanese waste--and the latter is considered to be a very unlikely resolution.

Draft, Section 3.6.6.1Issue

This section is labeled "Site Selection and Preparation" but nothing is mentioned of site preparation. What is involved in preparing a seabed site for use? (208-NRC)

Response

Site preparation would include the placement of locating devices on the bottom in such a way that the position of the ship and the canister in question will be continuously known. Preparation also includes a complete survey in three dimensions of the area of interest. This preparation is well within present state-of-the-art, deep ocean technology.

Draft p. 3.6.1Issue

The fourth paragraph states that a ship will monitor the emplaced wastes for an "appropriate length of time." How long (or short) is this "appropriate length of time?" (113-EPA)

Response

This period will be determined on the basis of engineering feasibility studies and model calculations of the subseabed system. It is anticipated that an appropriate length of time would be that time necessary to provide assurance that the waste package is properly emplaced in the sediments, that effective entry hole closure has occurred, and that the package temperatures are in accordance with prescribed operating limits. Package

ALTERNATIVE DISPOSAL CONCEPTS

emplacement would not occur on a routine basis until a waste disposal site had been qualified through emplacement of and monitoring of a number of test packages. Monitoring of test packages would likely continue after routine operations had begun.

Draft p. 3.6.1Issue

It is stated that the goal "to aid in solving national and international legal and political problems" will be started only after the technical and environmental feasibility is demonstrated. Has this been factored into the schedule that has been developed for this program? What lead time and resources have been planned? Has the DOE participated in any international discussions of this problem. A description of the programs of other countries interested in seabed disposal would be helpful. (208-NRC)

Response

A discussion of international legal and institutional considerations is included in the final Statement Section 6.1.4 . This section mentions factoring these considerations into scheduling as "it is not too early" to identify them. (See also Sandia 1980b)

Draft p. 3.6.1Issue

The first paragraph of this section should state that at present it is illegal to put high-level wastes in, on, or under the seabed and that legislative action would be required before implementation. (113-EPA)

Response

There are various differing positions at this time as to the legality of subseabed disposal or legislative action required. During the next few years the subseabed program includes planning for research at universities and institutes within the U.S. to identify these issues and attempt to resolve them.

A discussion of the legal and political concerns has been revised in the final Statement and is presented in final Section 6.1.4.4.

Draft pp. 3.6.1, 3.6.4Issue

A surface current gyre is a partially closed circulatory (not circular) system of surface and upper layer waters. (23-DOC)

ALTERNATIVE DISPOSAL CONCEPTSResponse

The comment is valid and appropriate changes were made in the final Statement.

Draft p. 3.6.2Issue

The difficulty of documenting a repository's location for future generations is presented as a major disadvantage of the seabed concept. Explain why this would be any more difficult to do for seabed than for conventional geologic disposal? (208-NRC)

Response

Documenting disposal locations in the subseabed concept is now seen to be no more difficult than for continental geologic disposal, and is well within the capabilities of current oceanographic technology.

Draft p. 3.6.2Issue

The last bullet under "Advantages" suggests that an advantage to seabed disposal is the lack of need to resolve Federal-State relations problems. This is not so because problems would surely arise from port use and the loading and transportation of waste to the port. (113-EPA)

Response

There is a difference between the Federal and state problems related to a transient (30-year) dock facility where no waste will be disposed of, and Federal and state problems relating to a disposal facility where wastes will remain forever and the chance of contamination may be seen by the local public to be higher. The commenter is correct, however, in that there will be Federal and state problems relating to location of a dock facility and to train or barge transportation to that facility.

Draft p. 3.6.2Issue

In mid gyre there is little benefit from deposition of sediments since this process is very slow there. The document states that less than 0.01 percent of the ocean floor would be used for disposal. What total area does this represent? (113-EPA)

ALTERNATIVE DISPOSAL CONCEPTS

Response

No benefit is claimed from additional deposition after canister emplacement. For the total area needed on the ocean floor, see Emplacement, Section 6.1.4.2.b of the final Statement.

Draft p. 3.6.2

Issue -

Section 3.6.1 discussed advantages and disadvantages. Suggest adding consideration of benefits of the low temperature and large heat sink resources automatically a part of the subseabed environment. (124)

Response

The benefits of the low temperatures and the large heat-sink capabilities of the ocean are being considered. The relatively small distance from the canister to the heat sink is considered an advantage.

Draft p. 3.6.3

Issue

Two--MPG1 and MPG2--are named but do not seem to come up again in the discussion. A map showing the areas, or at least listing the coordinates, would be a help. (124)

Two study areas were identified as having been chosen in the central North Pacific. Where are these study areas located? (Locate on a map.) (208-NRC)

Response

MPG areas 1 and 2 are in essentially the same location, at about 30°N, 158°W.

Draft p. 3.6.3

Issue

The Statement--"This region (the continental margin) is therefore unsuitable for consideration as a possible waste disposal site."--is too final for such a large region and cannot be justified without detailed discussion. A much more reasonable and specific statement is that made for fracture zones in the mid-ocean ridge: "On the basis of present knowledge, therefore, the fracture zones are not probable candidates as study sites."

Similarly the statement, "The abyssal plains . . . are therefore unacceptable for further consideration." should be modified. (208-NRC)

ALTERNATIVE DISPOSAL CONCEPTSResponse

The continental margins have been removed from consideration as possible sites for waste disposal. Reasons for this include present knowledge of the abundance of biological and mineral resources in these areas; the frequency and broad use of coastal waters; the relatively low and highly variable sorptivity of coastal margin sediments; and the difficulty in developing reliable methodologies for predicting the future characteristics of sediments in these regions. Similarly, proximal (i.e. landward) portions of abyssal plains, where relatively thick and permeable sand layers generally exist, have been removed from consideration. (See also Talbert 1979 and Talbert 1980.)

Draft p. 3.6.3Issue

The footnote on p. 3.6.3 discusses abyssal "deserts". This may be an overstatement. As abyssal depths continue to be explored, we are finding that deep areas are not biologically sterile. We know the real desert is a complex and fragile ecosystem. The lay reader, however, may interpret desert to be equal to sterile. Table 3.6.1 is more accurate and corrects the overstatement in the footnote. Suggest deleting the note entirely. (124)

Response

The deep sea is likely to be equivalent to the desert in terms of being a subtly complex ecosystem. While the deep sea has a very low standing crop, "desert" is not an accurate simile. As suggested in this comment, the note has been deleted from the text of the final Statement.

Draft p. 3.6.4Issue

The sediment thickness is reported to be 50 to 100 meters, while in Table 3.6.1 it is given as 100 to 300 meters. (208-NRC)

Response

Sediment thickness depends upon many processes: erosion and deposition, size of the ocean basin, inputs from the continental margins, etc. Generally, the older the crust, the thicker the sediments. The sediments have a thickness of approximately zero at the midocean ridge and get progressively thicker as one moves away from the ridge across the older plates. Sediments vary in thickness from ocean to ocean as well as within each ocean. The minimum thickness presently thought to be needed (from basement rock to sediment surface) is 60 m, but this is tentative, and depends on what is learned from the ion-transport studies that are just beginning to produce results.

ALTERNATIVE DISPOSAL CONCEPTSDraft p. 3.6.4Issue

A statement is made regarding waste disposal in trenches: ". . . a plate being subducted would have moved only tens of kilometers during that time (250 to 500 thousand years) and would not be subducted fast enough for waste disposal purposes." This conclusion does not follow from the discussion proceeding it in the same paragraph.

- a. How far would the waste have to move during that time to be subducted fast enough for waste disposal purposes? Reference?
- b. What might the impact be of the waste not being subducted fast enough? (208-NRC)

Response

In general terms the first several hundred years are the most hazardous. During that period, the plate would have moved less than 200 m, which for practical purposes means it has not moved at all. The water column in a trench is dynamic and unpredictable, and contains many complex processes such as turbidity currents and landslides in addition to those we know of in the open ocean. In other words, the most unpredictable regimes in the oceans are the trenches, and for that reason have been removed from consideration for the purpose of the Statement. Responding to the specific comments:

- a. The waste would have to be moved on the order of 100's of kilometers in order to be subducted.
- b. Should the waste not be subducted fast enough, it could become part of the over-riding plate and end up on the continental slope.

Draft p. 3.6.4Issue

"The time needed to contain the waste (250 to 500 thousand years)," shows again a lack of understanding of the role of containment and isolation." (154)

Response

To establish a "target" time of containment and isolation, the longest half-life of the waste constituents is chosen--that of Pu (25,000 yrs)--and multiplied by 10, which yields 250,000 to 500,000 years. The waste will contain many kinds of nuclides with widely varying half-lives, and ultimately DOE intends to separately address each nuclide, its half-life, and its necessary containment time to meet the pertinent criteria.

ALTERNATIVE DISPOSAL CONCEPTSDraft p. 3.6.4Issue

It is noted here that containment is required for 250,000 to 500,000 years. This appears inconsistent with current proposed criteria, which would require that the wastes maintain their integrity until they reach a radioactivity level comparable to ore bodies. This would entail about 500 to 600 years of storage, not 500,000. (124)

Response

In establishing early criteria, ultra conservatism was used. It is felt that any radioactive material entering the water column should have lower radioactivity than natural ore bodies, and therefore a time much longer than 500 years is needed. The figure of 500,000 years may, however, be too large.

Draft pp. 3.6.4, 3.6.7Issue

Under "Water Column," there is a statement that bottom currents are slow and uniform. However, in Section 1.3.6 the DEIS says bottom currents are weak and variable. This inconsistency should be corrected. (113-EPA)

It is stated that: 'Bottom currents in the MPG areas of the North Pacific are generally weak and variable.' A reference should be provided. How weak and variable bottom currents affect emplacement, radionuclide migration, heat transfer, etc., should be discussed. (208-NRC)

The bottom flow is not uniform and plate-like. The mean flow is thought to be slow, but there is a transient component that is related principally to the tides. Within about 5 m of the bottom there is a boundary layer that is thought dynamically analogous to the lowest 2 km in the atmosphere. See Weinbush, M. and W. Muck (1970): The Benthic Layer in The Sea, Vol. 4, pt. 1, A. E. Maxwell, Editor. (23-DOC)

Response

The discussion on water columns has been revised and condensed for the final Statement; the inconsistency noted above no longer appears. Bottom currents are weak and variable. (See also Bishop 1975, and Talbert 1976.)

Draft p. 3.6.5Issue

It is noted that the waste should decay to innocuous levels. It would appear appropriate to define innocuous levels. (124)

ALTERNATIVE DISPOSAL CONCEPTSResponse

"Innocuous levels" are tentatively defined as the levels that would be reached when a waste package had been allowed to decay for 10 half-lives for any radionuclide of interest. NRC and EPA are mandated by Congress to define "innocuous levels", and the definition employed in the Statement must remain nebulous until they have defined the term.

Draft 3.6.5Issue

There should be a mention that the philosophy behind this approach is isolation of the waste. This approach is required by EPA regulations. (113-EPA)

Response

DOE agrees. Isolation is accomplished by multiple barriers.

Draft p. 3.6.5Issue

The biological productivity of seamounts should be included in the table. (113-EPA)

Response

Reasonable data on the biological productivity of seamounts was not found during preparation of this Statement. Seamounts have not been considered for reasons of geologic stability, uniformity, and predictability. Thus, additional biological data was deemed unnecessary.

Draft pp. 3.6.5-7Issues

The multibarrier concept discussion in Section 3.6.2.3 makes several points that need clarification:

The Near-Field Effects (Section 3.6.2.3.b) presentation fails to make clear why heating is a problem--if the canister merely sinks or the clay sediments only rise, the radioactivity is still contained. The serious risk is rupture of a canister and accelerating release or radioactivity by the heat driven movements.

The Canister (Section 3.6.2.3.f) is described here as only a temporary barrier. This statement should be highlighted, being crucial to the whole assessment.

- a. Thermal behavior, coupled with breached canisters, could lead to an upwelling of a (relatively) narrow radioactive plume in the ocean.

ALTERNATIVE DISPOSAL CONCEPTS

- b. The likelihood of biotransport is enhanced by the combination of thermal and solution effect. (124)

Response

If the canister sinks, it will approach bedrock which is fissured and allows high water migration rates. If the sediment rises and carries the canister upward, the sediment column and, hence, long-term barrier is rendered less effective.

The canister currently envisioned is a temporary barrier in the sense that it is intended to contain the waste only through the heat generation period after which a breached canister will release wastes at essentially ambient temperatures only. It should be emphasized that the penetrometer represents an additional barrier to waste release, and that additional engineered barriers might be incorporated into the package.

Draft p. 3.6.6Issue

Supposing the correctness of the assumptions, the breakthrough time, T, is not a million years, but is:

$$T = D^2/A = (100 \text{ m})^2 / (3 \times 10^{-6} \text{ cm}^2/\text{yr}) = 3 \times 10^{13} \text{ yrs} \quad (23\text{-DOC})$$

Response

The A value should be:

$$3 \times 10^6 \text{ cm}^2/\text{sec} = 3.15 \times 10^7 \text{ sec/yr} \times 3 \times 10^{-6} \text{ cm}^2/\text{sec} = 9.5 \times 10 \text{ cm}^2/\text{yr}$$

Thus:

$$D^2/A = 10^8 \text{ cm}^2 / 9.5 \times 10 \text{ cm}^2/\text{yr} = 1.05 \times 10^6 \text{ yr}$$

or approximately 1 million years.

Draft p. 3.6.6, Table 3.6.2Issue

The sorption coefficients listed are credible although they could easily be 1 or 2 orders of magnitude in error. Their source should be properly explained and cited. They should not be used for transport-time calculation. (218-DOI)

Response

The sorption coefficients have been measured and their values have been documented. (See Russo 1979, Talbert 1977, Talbert 1979, Talbert 1980, and McVey 1980)

The transport time calculation is based on an assumed molecular diffusion constant wherein no interaction between the dissolved nuclides and the sediment is assumed.

ALTERNATIVE DISPOSAL CONCEPTSDraft p. 3.6.6Issue

"A pore water velocity of 10^{-1} m/yr (a factor of 10^5 over natural pore-water velocity)," reflects the hypothetical hyperbole syndrome again. (154)

Response

The reason that the calculations were so conservative is that the mass transport code which was used was not capable of using pore water velocities smaller than 10^{-1} m/yr. Even with these higher velocities, however, a water molecule did not move far, and would have been contained within the near sedimentary geologic formation.

Draft 3.6.7Issue

Under "Basement Rocks" the fracture nature of the basalts could provide lenses for the transport of radionuclides. This should be mentioned. (113-EPA)

Response

The basement rocks are indeed severely fractured, furnishing avenues through which nuclides could be transported more rapidly than through sediments. For this reason an equal thickness of sediments above and below the canister, a minimum of about 30 meters between the can and the surface, and an equal distance between the can and the basement rocks are being sought.

Draft p. 3.6.7Issue

Delete estimates of barrier properties, or add "if any" to read: to allow estimates of barrier properties, if any, of the water column . . . (23-DOC)

Response

The water column is not considered a containment barrier. However, it is a limited dilutional barrier. It is a large barrier to man's intrusion. Only a nation or other civilization with high technology has the capability to reach the site.

ALTERNATIVE DISPOSAL CONCEPTSDraft p. 3.6.7Issue

Why would one expect low nuclide concentrations around a waste canister? If the canister failed, one would expect high levels. (113-EPA)

Response

Canisters will be designed so that containment rather than failure will be expected. In the multiple-barrier concept, there is a waste form inside the canister which, if the canister failed, would release nuclides very slowly if at all, and therefore one would not expect high contamination levels.

Draft p. 3.6.7Issue

Previous reports on the U.S. seabed disposal program have not included the water column as a design barrier. Is it the program's intention to now identify the water column as a primary design barrier to radionuclide migration, or rather to investigate its properties as a barrier only for unexpected releases? In other words, do the conceptual plans allow for radionuclides to enter the water column during the period when they may present a hazard to man or the ecosystem? What is meant by inadvertent release? Scenarios leading to inadvertent release should be described. (208-NRC)

Response

The water column, which extends from the benthic boundary layer to the surface of the water, would only provide dilutional mitigation to the release of radionuclides. It is not considered an isolation barrier between the water and the biosphere, but is considered a barrier in the sense that it impedes man's intrusion into the seabed repository.

"Inadvertent" means "unintended by man". It is not appropriate at this early stage of the program to develop all possible scenarios of inadvertency.

Draft p. 3.6.7Issue

The fourth sentence of paragraph 6 appears to overdramatize the potential effect of heating and should be more realistically reworded. Also, the "risks" associated with sinking of the waste in the sediment should be explained, particularly in light of the preference for penetrometer emplacement. (58)

ALTERNATIVE DISPOSAL CONCEPTSResponse

A list of major problems has been identified as needing resolution before technical and environmental feasibility can be demonstrated. Of those that remain to be addressed, one is the movement of the canister or the sediment due to heat after emplacement. It will be two more years before assessment of the risks associated with either the sinking of the waste can through the sediment or the rising of the sediment and canister can be made. In either case, however, the risks attendant upon such movement will be minimal unless the canister is breached, which should not happen until at least 500 years has passed.

Draft p. 3.6.8Issue

Under current EPA regulations the canister must act as a barrier until the material decays to innocuous levels. The conservative calculational assumption, that the canister will release its entire inventory of wastes, does not reflect this regulatory requirement. (113-EPA)

Response

Although the canister will be engineered to contain the waste during its initial period of high activity, calculations based on instantaneous release of the entire canister inventory of radionuclides provide the upper limit for the rate of release.

Draft 3.6.8Issue

"Since repackaged spent fuel rods contain less (emphasis added) fissionable material and fewer fission fragments than does a similar volume of processed HLW." This is true for fission fragments, it is not true for fissionable material.

"Sediments which are hot (over 200°C) and moist." Nowhere can we find any value for the thermal conductivity of these "moist" sediments. It would seem as though the value might be higher than for conventional geologic media being considered. If so, the surface temperature of the canisters should be lower than that otherwise expected in conventional disposal systems. (154)

Response

The discussion in the draft Statement addresses the thermal impact of the waste. The higher fissionable element but lesser high heat fission fragment content of spent fuel produces much less heat in the first several hundred years than the fission fragments of HLW. The thermal conductivity of saturated sediments at temperatures of 200°C and pressures of 500 bars in the subject of current study. (See also Sandia 1980a.)

ALTERNATIVE DISPOSAL CONCEPTS

Draft p. 3.6.8, 9

Issue

The Emplacement Section (3.6.2.4) is incomplete and inaccurate.

- The implication is that any drillship or trenching concept would require a second ship and that no technology is known for doing both hole drilling and emplacement from one ship. Drillships can certainly do both.
- The section also implies that the penetrometer is simple and accurate. Such is not the case. It could act like an unguided glider slipping its way down into the seabed. It may soft-land, ricochet off the bottom, or enter at an angle. As a result, its position may not be known, it may not be retrievable, and it can not be considered an accurate, simple way to emplace wastes.

The retrievability discussion (3.6.2.6) is misleading. It implies that the penetrometer is easiest, and hole-drilling, the hardest. As a matter of fact, GLOMAR CHALLENGER has demonstrated time and again in the Deep Sea Drilling Program of the National Science Foundation the ability to reenter a drill hole of about 18" in diameter at a depth of 10,000 feet. On the other hand, retrieval of penetrometer emplaced items has not been accomplished in situ.

Ocean Transportation (Section 3.6.4.5) needs more thought. The docking facility, not described, is supposed to be able to handle three ships, each operating on a schedule of four trips per year. Given the realities of dock utilization, it allows each ship only about ten days per trip at the dock. There will be a lot of times when a second ship will be waiting at anchorage for dock space. This seems a bad idea for ships handling this kind of material and performing such specialized tasks. (124)

Response

Drillships can certainly be used both for transport, drilling, and emplacement. But it would not be efficient to use a drillship for transport, in which most of its time is spent going to and from the dock facility.

The penetrometer is not a "glider", making soft landings or ricocheting off the bottom. With a nose and tail fitted, it is guided into the bottom at a predetermined location and to a predetermined depth. Its position would be known from its time of release until it came to rest because it would be tracked by a transponder array on the bottom. Furthermore, present technology allows accurate position determination.

There is no implication that the penetrometer is easiest and drilling the hardest: it is stated that the penetrometer is cheapest, and drilling the most expensive. Both are established facts, having been demonstrated with scale models in the deep ocean. The only remaining questions relate to the size of a penetrometer that can handle the waste, or the size of the drill that can make a hole big enough to hold a canister one foot in diameter.

ALTERNATIVE DISPOSAL CONCEPTS

The commenter misinterpreted DOE's information about ships at the dock facility. It is desirable to have ships waiting empty at the dock so that when filling starts it can be completed rapidly, and the ship can immediately move to the disposal area and be emptied as soon as possible. Under these circumstances, 10 days at the loading dock seems reasonable. The only suboptimum factor is that having a ship at the dock waiting is not as economical as if it could load immediately upon arrival.

Draft pp 3.6.10, 22, 23, 31

Issue

Seabed disposal refers to a disposition of wastes and as such falls within EPA regulatory authority for the disposal of radioactive waste in, on, or beneath the ocean floor. The seabed disposal option for HLW is not legal under current domestic law. However, we think DOE should continue to study this option to see if this is an environmentally acceptable option. (113-EPA)

Additionally, in Section 3.6.3.6, on p. 3.6.22, the DEIS states "implementation of a sub-seabed disposal program for non-HLW is now possible under EPA's ocean disposal permit program." DOE apparently believes dumping and sub-seabed emplacement are intrinsically different for high-level waste and identical for low-level waste. We believe there is no legal difference between ocean dumping and sub-seabed emplacement and any difference between the two is purely semantic. (113-EPA)

The fourth paragraph of this section perpetuates the notion that sub-seabed emplacement is not ocean dumping. We consider the difference between the two to be semantic. (113-EPA)

Again we find the semantic difference between ocean dumping and sub-seabed emplacement. Dumping and Dump should not be in quotation marks. It is defined in the Marine Protection, Research, and Sanctuaries Act of 1972 as a disposition of materials. This misleading section should be corrected in the Final EIS. (113-EPA)

Response

For the purpose of this discussion, the DOE provides distinction between ocean dumping and subseabed emplacement to draw attention to an important factor of the subseabed concept. The subseabed concept relies on the isolation characteristics of the sediments to act as an effective barrier between the waste and the accessible environment. As such, careful emplacement into the sediments in a relatively well defined location of known characteristics followed by emplacement hole closure is essential. Conversely, ocean dumping is often used to connote an action of discarding without consideration for prolonged separation from the water column and without a proper characterization of the site.

ALTERNATIVE DISPOSAL CONCEPTS

While final legal interpretation of the existing statutes will be required to resolve the point, it is also considered possible that international agreements could be generated to allow subseabed disposal, if technical merit and minimal environmental impact can be demonstrated.

Draft p. 3.6.14

Issue

This statement requires documenting evidence or a reference. (218-DOI)

Response

The referenced statement "The benthic regions below such areas are marine deserts more devoid of life than the Sahara" has been replaced by a less restrictive statement in Section 6.1.4.5 (see Potential Events) of the final Statement. However the quotation in the draft Statement originally was derived from Talbert 1977.

Draft p. 3.6.14

Issue

Bottom sediments in the mid-plate areas have extensive animal tracks. Furthermore, fish in these areas make extensive vertical and lateral migrations; this indicates that there is a possible pathway from the waste to people. (113-EPA)

Response

Over the next 15 years, the determination of the impacts of the water column and of biological movement on the transport of radionuclides from an accident back to man is planned to be made.

Draft pp 3.6.16-18

Issue

"d. Predictability" The first paragraph is particularly appropriate to all concepts and media. The entire section is generally applicable to all geologic disposal concepts, and the subject should be addressed for all host media, not just this one. (154)

Response

It is agreed that the paragraph (as opposed to the entire section) is generally applicable to all concepts and media. Performance assessment capability to a very high degree must be developed and expressed on all concepts.

ALTERNATIVE DISPOSAL CONCEPTSDraft p. 3.6.18Issue

Appropriate references on these sorption coefficients need to be cited here. (218-DOI)

Response

These conditions are specified for the experiments, and the environmental conditions deep within the sediments are also being characterized at this time. (See Talbert 1979 and Talbert 1980).

Draft p. 3.6.19Issue

These sorption coefficients mean very little unless their exact measurement conditions (chemical and physical) are described. This would include as a minimum: ionic strength and composition of the solution, initial exchange conditions of the clay, pH and Eh of the solution, surface area of the clay, sorption equilibrium time, and solid-to-liquid ratio. (218-DOI)

Response

This figure summarizes data presented in the 1975 Seabed Disposal Program Annual Report (Talbert 1976).

Draft p. 3.6.19Issue

This statement appears in error. ^{54}Mn shows more than a factor of 10 spread. (218-DOI)

Response

^{54}Mn does show slightly more than a factor of 10 spread; however, ion transport calculations show such conservatism that a difference covering two orders of magnitude would still be a safe and acceptable level. As pointed out, sorption characteristics are probably less important than permeability.

ALTERNATIVE DISPOSAL CONCEPTS

Draft, p. 3.6.19

Issue

Although organic complexes might be insignificant, there is a chance that carbonate (or other inorganic) complexing might be significant. This should be pointed out. (218-DOI)

Response

Inorganic complexing could indeed be a problem, but the complexing probably will come from leaching of the waste form rather than from the sediments. Carbonate complexes would be less likely since the planned water depth is well below the calcium carbonate compensation depth; therefore, there is very little carbonate available for complexing.

Draft p. 3.6.20, line 10 from bottom

Issue

Replace "a poor" with "no" in the sentence beginning on line 20 so that it reads: For the reasons given, the water column is likely to be no barrier against large quantities of nuclides.... (23-DOC)

Response

The water column is not considered a primary barrier. However, it will inhibit man from intruding, and it can contribute both isotopic, volumetric dilutional and dispersional benefits.

Draft p. 3.6.20

Issue

Under the discussion of the water column, it should be recognized that while the water column may not provide a barrier to migration, its enormous capability to dilute such releases below significant concentrations cannot be overlooked as a mitigative feature. (208-NRC)

Response

The point is being taken into account. However, DOE considers it of secondary importance to the primary barriers (canister, waste form, and geologic formation).

ALTERNATIVE DISPOSAL CONCEPTSDraft p. 3.6.21Issue

The research and development costs to support the penetrometer emplacement concept are quoted as \$250 million on p. 3.6.21, and as \$60 million on p. 3.6.31. The components of each figure should be given. What is the meaning of "state-of-the-art" (Figure 3.6.1) referring to penetrometer emplacement, given the quarter of a billion dollar research and development cost estimate? (208-NRC)

Response

The presentation of costs for this alternative has been improved in the final Statement. R&D costs estimated for the subseabed program are \$250 million. A reliable estimate of R&D costs for the development of the waste package, including penetrator, is not available.

Draft p. 3.6.21Issue

The basis for the following cost estimates should be provided (including the components and assumptions for each):

- a. "The resulting order-of-magnitude figure is \$200 million for the capital cost of handling 1800-3600 MTHM/hr" (p.3.6.21).
- b. The \$25 million/year operating cost (p. 3.6.21).
- c. "It is estimated that the program can be completed in 25 years at an overall cost of about \$560 million including construction of one ship and a port facility" (p. 2.6.27). Details on the 25 year schedules should be provided.
- d. Each of the estimated costs of the multibarrier research and development program (Section 2.6.6.2). (208-NRC)

Response

Cost estimates have been improved and bases have been provided in the final Statement. For details on the 25-year schedule, see The Subseabed Disposal Program Plan, Volume I, Overview. (Sandia 1980a)

ALTERNATIVE DISPOSAL CONCEPTS

Draft p. 3.6.22-28

Issue

See comments for 3.6.10, Section 3.6.2.8. Subseabed emplacement must comply with EPA regulations promulgated under authority given exclusively to EPA under Public Law 92-532, the Marine Protection, Research and Sanctuaries Act of 1972. (113-EPA)

Response

Prior to application for a permit to dispose of waste in the subseabed sediments it would be necessary to successfully complete several procedural steps. These would likely include (in addition to current efforts to develop necessary information and verify environmental and engineering feasibility) a licensing and public review process as well as an EPA permit.

Draft p. 3.6.23

Issue

Should high-level waste be released, it most certainly will affect other nations, contrary to the suggestions in the fifth paragraph. (113-EPA)

Response

It appears that the third paragraph is actually being discussed. No similar opinion appears in the final Statement, as it is recognized that radionuclides which might be released from any disposal facility might have wide spread effects. The content of the original thought was to express the remote nature from populated land areas.

Draft p. 3.6.24

Issue

It should be made clear that tsunamis could pose no danger to a ship that was not in shallow, near shore waters, or near the source of tsunami. Even a large tsunami would probably not be noticed by a ship in mid-ocean because of the long wave length (typically hundreds of kilometers) and relatively small oceanic wave heights (usually less than a meter). A minor storm or just rough seas would pose greater danger in mid-ocean. (208-NRC)

Response

Tsunamis would pose no difficulty for ship directed emplacement operations or ship transportation to the disposal area. Tsunamis would only be of concern for water transport activities occurring in coastal regions.

ALTERNATIVE DISPOSAL CONCEPTSDraft p. 3.6.24Issue

Port accidents occurred in the 60's during the loading of 55 gallon drums. This issue should be presented. (113-EPA)

Response

It is not believed pertinent to include such a discussion in the final Statement. The practice of discarding low-level waste in 55-gallon drums has been discontinued by the U.S. due to this and other inherent problems. The handling techniques were routine industrial practice, and not consistent with those which would be instituted in a conservative nuclear waste disposal system.

Draft p. 3.6.24Issue

The Risks and Impacts section on Land and Sea Transport (Section 3.6.5.1) makes some incorrect assumptions, based on lack of review of actual ship accident data. Although intuition suggests operating in areas of low traffic volume will reduce accidents, data show that accidents are strictly proportional only to the number of port calls made. Thus, sailing on routes that avoid traffic does no good. Similarly, double hulls are not an accident-prevention measure, but a way of reducing the chance of an oil spill from tanker accidents. Since the radioactive waste will be in some kind of canisters, double hulls are of questionable value. (124)

Response

Sandia Laboratories has a program with the Department of Transportation that will address the location and frequency of ship accidents on the basis of study results; it will guide the program toward the most safe and reliable solution to the transportation problem. Little effort has been placed on ocean transport up till now because it is of greater importance to assess the technical, environmental, and engineering feasibility of the concept before major monetary outlays are made for transportation studies.

DOE agrees that double hulls do not prevent accidents. However, they do decrease the chance of a ship sinking or the cargo (waste canisters) being damaged as the result of a collision.

ALTERNATIVE DISPOSAL CONCEPTSDraft p. 3.6.24Issue

Is it proposed that several canisters go into one hole in the seabed, or will each penetrometer drop wherever it may, to be followed by monitoring of 9000 different holes per year? (113-EPA)

Response

If penetrometers are used, only one will be emplaced per location. If emplacement is by drilling, the number of canisters to be placed in each hole will be determined by the total thermal load allowed per unit horizontal area. In either case, monitoring will be of a region covering several canisters rather than of individual canisters.

Draft p. 3.6.25Issue

Several ports have banned the shipment or receipt of spent fuel. Does the proposal include use of dedicated port facilities? (113-EPA)

Response

It is assumed that the program will utilize a military facility on one or both coasts. This facility would be dedicated for the time of disposal of the wastes.

Draft p. 3.6.26Issue

Consideration of meteorite strike is truly incomprehensible. (154)

Response

It is agreed that the probability of meteorite strike is so small as to be almost incomprehensible. The discussion is included in the final statement, however, primarily because a similar scenario is evaluated for the mined repository, to which the subseabed concept must ultimately be compared.

Draft p. 3.6.26Issue

What is the range of error in the results of the "unverified, theoretical model" in projecting impact? (113-EPA)

ALTERNATIVE DISPOSAL CONCEPTSResponse

The range of errors will not be determined until the systems analyses are completed and the models are verified.

Draft p. 3.6.26Issue

The assumption of 0.5 MT of HLW per canister is appreciably lower than 2.5 to 3.0 MT usually assumed for HLW canisters and should be explained. (58)

Response

The assumption of 0.5 MT of HLW per canister is based on preliminary results from studies of the near-field sediments and canister corrosion, which indicate that canister interface temperatures of approximately 200-250°C should not be exceeded. Reasons are: (1) sediment mineralogic and flow characteristics are not appreciably altered at these temperatures, and (2) it is postulated from preliminary corrosion data that a canister capable of 500-year integrity can be obtained if the temperature remains below 250°C.

Draft p. 3.6.28Issue

"...identifying centimeter-size imperfections in the sediment surface and decimetersize imperfections at depth."

On what basis were these quantitative criteria developed? Why are such criteria only here and not defined for imperfections in the host media of other concepts? (154)

Response

The final criteria for subseabed siting has not been developed and it is unlikely that identification of imperfections of this size will be required, even if the capability exists, unless a definite need is identified. As far as the need for similar criteria in other concepts, no obvious requirement exists.

Draft p. 3.6.29Issue

Measurements of molecular diffusion coefficients will be needed for near and far-field transport analysis. (218-DOI)

ALTERNATIVE DISPOSAL CONCEPTSResponse

This is true. Current efforts focus on the use of formation factors to correct free seawater molecular diffusion coefficients to in situ values. Site-specific measurements will be made as potential disposal sites are identified.

Draft p. 3.6.30Issue

Is the \$15 million mentioned for R&D costs? If not, what costs does that figure represent? (113-EPA)

Response

The \$15 million is for R&D costs only.

Draft p. 3.6.32Issue

The sixth option should be clarified. Considering the dollar input, what is the intent and what will be the output? (113-EPA)

Response

The intent of the sixth study area is to promote international cooperation in providing for assurance of the environmentally acceptable use of the oceans and subocean geologies for isolation of radioactive waste. The cost for this activity is small when compared to the international cooperative benefits to the U.S. in providing for assurance of acceptable methods for disposal of radioactive waste.

Ice Sheet DisposalDraft p. 1.27 and 1.37Issue

It should be clarified that we are waiting for development of disposal techniques not development of ice sheets. (34)

Response

The wording of this statement has been corrected in the final Statement.

ALTERNATIVE DISPOSAL CONCEPTSDraft Section 3.7Issue

Ice Sheet Disposal unacceptable because of the possible interactions between waste, ice sheets and weather variations. (88, 121)

Response

The ice sheet disposal concept is not considered a preferred alternative for the disposal of radioactive waste. One reason is that there are significant environmental impact uncertainties associated with this concept. Section 6.1.5.3 of the final Statement addresses the areas of uncertainty. Section 6.2 presents a comparison of alternatives that further discusses the importance of these uncertainties.

Draft Section 3.7Issue

Ice Sheet disposal - This entire section should be rewritten. Additional data from the Dry Valley Drilling Project, and the Ross Ice Shelf Project provide significant additional geologic scenario. (43)

Response

Alternate concept sections presented in the draft Statement have been significantly revised for the final Statement (see Chapter 6.0). The final Statement presents environmental and other information believed to be sufficient for comparison of the alternatives.

Draft Section 3.7Issue

The general overall caution and concern voiced in this section about using the ice sheet method is concurred with. This section (3.7) is better presented than 3.6 or 3.5 because it treats the uncertainties in a clearer fashion. (124)

Response

In the preparation of Section 6.1 of the final Statement effort was made to present the concepts in an equivalent and more comparable manner. However, complete equivalence of presentation was not possible because of the underdeveloped status of many of the concepts.

ALTERNATIVE DISPOSAL CONCEPTSDraft Section 3.7Issue

This concept would provide a high degree of isolation (our terminology - make man's intrusion unlikely). However, the litany of unresolved technical and political problems make it clear this concept cannot possibly be considered for an early repository. Furthermore its cost would likely rule it out for even much later consideration. (154)

If the 1% heat load criterion is reasonable (draft p. 3.7.8) then it should be stated and the high area (7%) demand would rule out the Antarctic approach. This report requires better perspective to be useful. A clear statement should be made to rule out this alternative. (124)

The 10-30 years of extra lead time required for ice sheet disposal, as compared to other disposal options, will delay implementation of a publicly acceptable nuclear waste management system...(Section 3.7.5.4) should be highlighted, because it makes this option extremely unattractive to comparison to others whose technology is more developed. (124)

Response

Section 6.2 of the final Statement presents a comparison of the attributes of alternative concepts and the mined repository. This comparison employs the use of standards for judgment to evaluate the concepts for the purpose of identifying preferred concepts. In general, concepts that have significant technical and environmental performance uncertainties are lower in preference than the better defined concepts. The ice sheet concept is, therefore, one of the lesser preferred concepts.

Draft Section 3.7Issue

The section treats rather severe transportation problems in a rather causal manner and without appraisal of non-radiological hazards. A more thorough and complete treatment of transportation, including accident and non-radiological hazards, should be provided. (58)

Under Section 3.7.1.5, the risks, hazards, and impacts of transporting HLW over ice in polar climates should be presented. (208-NRC)

The quality of the reasoning contained in the DEIS might be called into question by the apparently serious consideration of rather bizarre and untested methods of transport and emplacement of the waste in the Antarctic ice-sheet. The horrendous logistics problems of the Antarctic repository are discussed at length with no apparent consideration of the need for recognition and quantification of non-radiological hazards. (58)

ALTERNATIVE DISPOSAL CONCEPTSResponse

It is acknowledged that the safe transportation of wastes to and over the Antarctic ice would pose formidable engineering design and operations problems. It is believed, however, that the uncertainties of the ice emplacement medium itself and the Antarctic environment should receive greater attention for this Statement. There is no intent to minimize the other important issues such as those presented by the commenters. See Section 6.2 of the final Statement for comparative discussion of transportation characteristics for the alternatives and the mined repository.

In addition, Section 4.5 of the Predisposal Systems Chapter discusses transportation alternatives (including sea transport) in a more detailed and thorough manner than in the draft Statement. Quantitative and qualitative assessment of impacts were presented whenever possible using the most recent data available.

Draft p. 3.7.3Issue

It is not clear why the example heat criterion is a factor of 20,000 less than that used for salt. (154)

Response

The criterion referred to was presented only for the purpose of providing a frame of reference regarding how heat load criteria might affect the land area required for disposal. The first sentence of the referenced paragraph of the draft Statement stated that additional study would be required to determine the heat load criteria for waste emplacement in the ice sheets.

Draft p. 3.7.3Issue

It is proposed that aircraft be utilized to ferry waste canisters to emplacement sites. This is judged to be a poor choice in that risks of severe air transportation accidents are much higher than for ground transportation modes. (124)

Response

Section 6.1.5.2 of the final Statement notes that over-ice transport of casks via land vehicles is the preferred alternative (as opposed to aircraft).

ALTERNATIVE DISPOSAL CONCEPTSDraft p. 3.7.4Issue

A small body of data have been advanced in the past few years of more recent local glaciations (alpine type) and flooding of the dry valleys. Glacial permafrost drift locally exceeds 300 meters in Taylor Valley. (43)

Response

The referenced information would be reviewed for development of the ice sheet concept.

Draft p. 3.7.4Issue

It is not clear why more is not made of the dry valleys since they seem to be more suitable than the ice itself. (124)

Response

The final Statement discusses the use of ice-free areas of polar regions called "dry valleys" as possible interim waste canister storage locals. A mined repository in dry valleys would be considered akin to the conventional geologic option and therefore subject to the same considerations.

Draft p. 3.7.7Issue

Transportation - How many tons per year are we talking about? The realistic shipping season is two-three months (more like two months). Can the ground transport system handle the projected volume? In recent years, about one aircraft accident per year has occurred. The safety records, although enviable for harsh environmental areas, are still not good enough for carrying large quantities of waste. (43)

Response

While it is agreed that the severity of air transportation accidents is usually greater than that of surface transport accidents, the risk of air accidents involving radioactive materials has been demonstrated to be much lower than that of surface transport accidents. Careful consideration of the modes of transport would be a necessary part of any program to safely dispose of radioactive waste. See Section 4.5 of the final Statement for additional discussion of transportation issues.

ALTERNATIVE DISPOSAL CONCEPTSDraft p. 3.7.9Issue

The cost figure (Table 3.7.1) seems to be low. Recent purchases of C-130's for polar work are expensive. Logistics support is extremely high. The present WSARP (NSF) program is about \$40 million per year to support about 1,000 men and women in the summer and about 40 in the winter. About 90 percent of the costs are in logistics, and less than 10 percent is useful science. The environmental impact of large scale technology in polar regions may be too much to pay. (43)

Response

Cost figures have been updated in the final Statement (see Section 6.1.5.6).

Before such a disposal option would be implemented, a thorough study of environmental impacts would be made to determine acceptability in terms of environmental "costs". Section 6.1.5.4 addresses environmental concerns for the concept.

Draft p. 3.7.10Issue

Disposal of wastes in an ice sheet would probably entail loss of control of spacing of waste containers. Flow velocities and perhaps even flow directions might change over long periods. Possible impacts should be evaluated. (218-DOI)

Response

Environmental impact uncertainties of ice sheet disposal are addressed in Section 6.1.5.5 of the final Statement. Impacts beyond those reported are not currently known due to incomplete development of the the ice sheet concept.

Draft p. 3.7.10Issue

This statement should mention that there is one more handling and transportation step in ice disposal than in subseabed disposal-- from the unloading dock in the ice disposal site. (218-DOI)

Response

Section 6.2 of the final Statement includes a comparison of alternative concepts including ice sheet disposal and subseabed disposal transportation systems. The statement also includes the above-mentioned leg under "transport and handling", Section 6.1.5.1.

ALTERNATIVE DISPOSAL CONCEPTSDraft p. 3.7.10Issue

This statement (first item) is debatable; the fact that these remote areas are relatively unexplored for resources might attract considerable exploration in the future as some of the last frontiers for new discoveries. (218-001)

Response

The text of the final Statement has been revised to incorporate consideration of potential future exploration of remote Antarctic areas.

Draft p. 3.7.10Issue

Elsewhere in the text sub-ice lakes are identified. Present hydrogeologic studies strongly suggest that the sub-ice lakes provide the groundwater for the discharges in the dry valleys. One drill hole by the Dry Valley Drilling Project (DVDP 13) identified upward moving groundwater at -16°C at a depth of 150 meters. The water appears to have moved through fractures in the crystalline bedrock. Preliminary heat flow studies by DVDP suggest high heat flow (equivalent to the basin and range province of the U.S.), and the possibility that uranium has been leached to a depth of 300 meters. (43)

Response

It is believed that the comment refers to the discussion of dry valleys presented in the draft Statement. If so, a repository located in bed rock caverns would be subject to the same site selection and site assessment criteria as those located in the continental U.S. Site hydrology would necessarily be completely characterized before a selection would be made. The referenced information and a great deal of additional hydrologic data would be included in the characterization.

Well Injection^(a)Draft p. 1.28Issue

Although grout injection technology is rather well established, long-term durability of grout seals is unknown. It appears that considerable effort is needed to develop grout types and sealing techniques that can be relied upon with confidence for many millenia. (218-001)

(a) This concept was called "Reverse Well Injection" in the draft Statement.

ALTERNATIVE DISPOSAL CONCEPTSResponse

The need for seal systems development is discussed in the final Statement in Section 6.1.6.3, Status of Technical Development and R&D Needs. Estimates are not currently available for cost or time to develop technology that would be necessary for the implementation of the well injection concept. See Section 6.2 of the final Statement for a comparative assessment of alternative waste disposal technologies.

Draft p. 1.28

Issue

What advantages does the reverse well disposal alternative have? It is not clear why it is being considered. (34, 154)

Response

The well injection concept involves pumping grout slurry wastes to depths from 1000m to 5000m into porous or fractured rock strata which would be isolated from the biosphere by relatively impermeable overlying strata. The grout mixture would harden to a low leach mass. This is a relatively low cost alternative concept designed to retard nuclide movement by minerals within the rock strata. Because of the depth of disposal, the probability of a breach by either natural or human induced events are low. See Section 6.1 of the final Statement for additional discussion of the well injection concept. Also see Section 6.2 of the final Statement for a discussion of the attributes and disadvantages of a alternative concepts and the mined geologic repository.

Draft pp. 1.28 and 1.32

Issue

The discussion and assessment of the reverse well injection alternative should include a general consideration and evaluation of the vertical separation of individual plates or sheets of injected waste-bearing grout. Consideration of the impacts of the alternative should also include at least generalized assessment of effects of possible deviation of sheets from bedding planes and possible unplanned or unscheduled accumulation of wastes in any given zone. (218-DOI)

Response

Consideration and evaluation of the vertical separation of individual plates or sheets of injected waste-bearing grout would be important in the design of a well injection disposal system. For example, the number of grout sheets and their separation would be dependent on the gross thermal loading acceptable for the host rock.

ALTERNATIVE DISPOSAL CONCEPTS

Such information is not currently available for HLW waste injection. Other information that is presented in the final Statement should be adequate, however, to provide a comparison of the technical alternatives for the disposal of HLW, spent fuel, contact handled TRU, and remote handled TRU wastes.

An example of the assessment of deviation of grout sheets from a horizontal is given in the Environmental Statement for the ORNL operation.

Draft p. 1.29

Issue

The total capital cost for grout injection facilities at all five disposal sites is estimated to be \$300 million with \$10 million for annual operating costs. To which fuel cycle does this apply? Also what type of wastes do 12 states prohibit from being injected into deep wells? (147)

Response

Section 6.1.6.6 of the final Statement contains a more up to date and accurate discussion of cost. The type of waste that "12 states prohibit from being injected into deep wells" is high level waste.

Draft Section 3.8

Issue

The report treats Shale Grout Injection as a form of Reverse Well Injection. The latter should be treated separately, since the shale fracture aspect and the waste fixation in grout represent very different forms of isolation than that for reverse well injection of liquid waste in porous formations. (11)

Response

In preparing the final Statement, the section on well injection (Section 6.1.6) was revised in order to more clearly delineate between the shale grout injection and the reverse well injection techniques.

Draft Section 3.8

Issue

In today's climate, and considering the exquisite detail with which the possibility of leaching of highly insoluble waste forms might occur was discussed in Section 3.1, it is inconceivable that either "reverse well" concept could possibly be acceptable politically in any foreseeable future. (154)

ALTERNATIVE DISPOSAL CONCEPTS

Well developed technology for this option is seriously offset by uncertainties regarding the ability to demonstrate satisfactory long-term isolation which is strongly dependent on the host geohydrological system. (124)

Response

A major disadvantage of the well injection concept is that the waste form is not compatible with the multibarrier concept which is considered important for technical conservation. Attributes and disadvantages of the alternate concepts are compared in Section 6.2 of the final Statement.

Draft Section 3.8

Issue

The section adequately enumerates the advantages, disadvantages, and potential problems that must be addressed if well injection is to be used as a method of radioactive waste disposal. There are several considerations which, although briefly mentioned in the report, realistically cast serious doubt on the entire concept of utilizing well injection as a safe method of radioactive waste disposal.

Beginning with the shale-grout method, the critical aspects are the control of the orientation of fractures in which the waste is implaced, the leachability of the shale-grout mixture and its stability over time in a groundwater environment, the relatively shallow depth at which the waste is stored and the problems in maintaining an undisturbed or unpenetrated geologic environment over long periods of time.

In isotopic homogenous model studies, control of hydrofracture orientation is accomplished in a relatively straight-forward manner. In a real geologic environment, anisotropy and inhomogeneity are the rule. In addition, existing fracture systems controlled by post depositional stress on the rock units and later tectonic forces are present in rock units from granite to poorly consolidated glacial-till. Those zones of weakness are difficult to detect in rock cores but will be the controlling factor in the orientation of artificially induced fracture systems, as important as the vertical and horizontal stress components discussed in Section 3.8.

The effects of existing fracture and joint systems should be addressed in a much more specific manner. It is probable that the presence of fracture systems will be found in any proposed repository zone and that their presence would be cause for the elimination of the shale-grout disposal method.

A further note here which is also applicable to the other following points of discussion is that in groundwater flow through shales of low permeability it is the fracture system which will control the amount of water flowing through the unit and not the low permeability of the shale itself.

ALTERNATIVE DISPOSAL CONCEPTS

This leads to the leachability and stability of the grout mixture. The binding agent is a combination of calcium carbonate and calcium silicate, both of which will be under-saturated in most flow systems encountered at the shallow depths required for this system. The stability of this binding agent should be addressed in more detail. It is not sufficient to rely on the presence of the shale to sorb any ions released by the dissolution of the cementing agent as most flow will be occurring in fractures created in the shale-grout mixture. (43)

Response

The effect of existing fractures and strength anisotropy when known can be included in an analysis of propagation directions. With multiple fracture sets, the combination of in situ stress and strength anisotropy will determine the fracture propagation direction. Depth limitations based on probable in situ stress conditions in the continental U.S. are discussed in the draft Statement in Section 3.8.1.4, and the final Statement in Section 6.1.6.2. Section 6.1.3, R&D Requirements, of the final Statement discusses the need for the collection of geologic data before the well injection concept could be implemented. The fracture permeability of shales is recognized as important (see Section 3.8.2.1 of the draft Statement)

Section 5.2.2.5 of the final Statement provides a discussion of requirements for depth of waste emplacement.

References 45 and 46 from the draft Section 3.8 contain data on the extensive testing undertaken by ORNL on grout leachability. Section 6.1.6.3 of the final Statement, R&D Requirements, briefly addresses waste material development needs.

Draft p. 3.8.1

Issue

A brief paragraph on retrievability appears. There is no assurance that the liquid waste, once pumped into a porous medium, is totally retrievable, a certain fraction of the waste will remain "captive" within the host rock. Total recovery, at any cost, is likely not attainable. A more detailed discussion focusing on the impact of partial recovery should appear. (208-NRC)

Response

The impacts of partial recovery are not yet fully known. Methods for effecting corrective action and the cost and benefits of their use would be a necessary development for the implementation of the well injection concept. See Section 6.2 in the final Statement for a comparative discussion of attributes including those pertaining to corrective action.

ALTERNATIVE DISPOSAL CONCEPTS

Draft p. 3.8.2

Issue

One suggested storage media is depleted hydrocarbon reservoirs. There are obvious problems with this, as additional hydrocarbon reservoirs are often found beneath depleted fields. Recovery from the underlying reservoirs would necessitate penetrating the liquid waste reservoir. As improved hydrocarbon recovery techniques are continually being developed, utilization of depleted hydrocarbon reservoir area storage medium may preclude recovery of otherwise-available natural resources. (208-NRC)

Are there any other examples of porous fractured strata that could be used for deep well injection that would give a more balanced treatment to this concept? (208-NRC)

Response

The use of hydrocarbon reservoirs was introduced as an example of present-day reverse well disposal which is typically used, for example, for oil field brines. However, it may be that hydrocarbon reservoirs would not be considered suitable for toxic or radioactive waste due to the numerous wells, some of which may be inadequately plugged for radioactive waste isolation, and the possible sterilization and prevention of future resource recovery. Deep well injection at the Rocky Mountain Arsenal was into a porous fractured rock. Details are given in Reference 37 of the final Statement.

Draft Section 3.8

Issue

Shale-grout wastes would be irretrievable--what about well contamination and other underground water currents? Ten million a year is an unacceptable burden to taxpayers. (88, 121)

Response

Leaching of the grout is recognized as the probable primary pathway to the biosphere. It would, therefore, require extensive laboratory and field testing to ensure that release rates are such that concentrations in the biosphere would be within acceptable limits.

A comparative assessment of the costs for the well injection is presented in Section 6.2 of the final Statement.

ALTERNATIVE DISPOSAL CONCEPTSDraft p. 3.8.9Issue

The discussion here of retrieval fails to address the potential reasons for retrieval or the net risk reduction that might be achieved. A more thorough treatment would probably indicate retrieval to be more harmful than beneficial. (58)

Response

Corrective actions for the well injection concept have not yet been developed. Identification of the risks and benefits of possible corrective actions would be a part of this development.

Draft p. 3.8.9Issue

The discussion on retrievability ought to address what fraction of the waste might be retrievable. It is difficult to see how the retrievable fraction could be much more than 0.5. (154)

Response

For well injected waste, retrievability would be species-dependent and thus would be likely higher for ¹³⁷Cs than for transuranics. It would depend to a great extent to the reactions which take place in-situ between the injected waste and the host medium. Current information is insufficient to estimate the fraction of each species that would be retrievable or the method by which such retrieval would be effected.

Draft p. 3.8.9Issue

The suggestion of acid flushing for retrievability is alarming. The unpredictability of solution mining is well known and acid flushing could never be justified in licensing procedures. (124)

Response

Reverse well disposal would generally be considered non-retrievable. The use of an acid as a flushing material for recovery would only be as a last resort. The use of acid flushing techniques might not be acceptable in licensing procedures, but it is mentioned in the final Statement for completeness.

ALTERNATIVE DISPOSAL CONCEPTSDraft p. 3.8.10Issue

How can this dismissal of the problems of isolation for these concepts be rationalized with the 50 or more pages of detailed consideration given to these possibilities in Section 3.1? The Final EIS could apply this assessment to geologic disposal, but undoubtedly will not. In which event it should recognize that these concepts are as susceptible to intrusion as conventional geologic disposal -- or more so. (154)

Response

In the preparation of Section 6.1 of the final Statement effort was made to provide balance in the discussion of alternative concepts. However, in many cases because of the relatively undeveloped nature of the alternatives the information presented cannot reflect the breadth and depth of consideration given to the mined geologic repository. Section 6.2 of the final Statement takes this into consideration in the presentation of comparisons of the alternatives and the mined geologic repository.

Draft p. 3.8.10Issue

The failure to distinguish between the concepts of containment and isolation (as we define them) is endemic to the entire Draft EIS but is more rampant than in these four pages. The two subsections labeled "Isolation" start with two and three bullets, respectively, which speak to the containment of waste. There then follows in each subsection the following statement:

"isolation formation unlikely to be susceptible to breaching the natural events (tectonism, volcanism, meteorite impact) or man-made events (resource investigation or exploitation, surface or subsurface activities, acts of war or sabotage)." (154)

Response

Please refer to the Glossary in Chapter 8.0 of the final Statement.

Draft p. 3.8.11Issue

Candidate Geologic Environments: Effects of local stresses and conditions -- such as varying topographic load and joint patterns -- on uplift resulting from reverse well injection should be further examined in the course of testing suitable monitoring methods that would not penetrate the individual waste sheets. (218-DOI)

ALTERNATIVE DISPOSAL CONCEPTSResponse

The use of uplift measurement in the area of grout injection is currently being used at ORNL to monitor performance.

Before the well injection concept could be implemented for HLW it would be essential that the total R&D program be supported by a data base that would cover all the components affecting performance of the disposal system. The effects of local stresses upon uplift resulting from well injection would likely be further examined.

Draft p. 3.8.13

Issue

It is indicated in the third sentence that the waste (emplaced by means of the reverse well option) is subject to rather simple unauthorized retrieval, however, Table 1.8 on p. 1.34 accords the reverse well option a 5 rating on susceptibility to short-term encroachment. The assessment of the withdrawal should be made more quantitative and the apparent discrepancy in characterization of the option should be clarified. (58)

Response

A discussion of retrievability is presented in the final Section 6.1.6.2. It provides a discussion of the retrievability of wastes and makes note of the fact (for liquid injection) that certain isotopes, for example ^{137}Cs , will likely remain in solution while others may be fixed by the host rock. Section 6.1.6.7 provides a discussion of safeguards issues important to the concept. Section 6.2 provides a comparative assessment of the alternatives in which corrective action, safeguards and future human intrusion, along with other factors are considered. Wastes that would be emplaced by well injection are considered irretrievable.

Draft p. 3.8.13

Issue

In the next sentence (at the top of page 3.8.13) the following is stated as a disadvantage of the concepts:

"The waste may progressively disperse and diffuse throughout the permeable host rock and eventually encompass a large area (volume?) although at a lower concentration and after decay has occurred."

This would be a good place to pick up and expand upon this 'equilibrium fringe release' concept. In the longer term what is needed is isolation and that reducing concentration is the ultimate in isolation. Thus this is an advantage for these concepts, not a disadvantage. (154)

ALTERNATIVE DISPOSAL CONCEPTSResponse

The final Statement states that the waste may progressively disperse and diffuse throughout the permeable host rock and eventually encompass a large volume, although at a lower concentration. The concentration may be variable and unpredictable. Thus there is no assurance that the waste would be diluted.

A discussion of currently accepted performance requirements for disposal of radioactive waste is presented in Section 6.2 of the final Statement.

Draft p. 3.8.19, 20

Issue

The following statement:

"Geology. The predictive capability of the geological sciences for basic parameters such as rock type, lithology, and structure is well developed using drilling, mapping, and geophysical techniques. However, minor variations in these parameters are of significance in deep well injection. Investigation and predictive methods for structural features (joints, faults) and lithologic or geochemical variations will require improvement."

stands in stark contrast to multiplicity of pages of geological discussion in Section 3.1. Again the treatment throughout the Final EIS should be more consistent. (154)

Similarly the following statement on sealing technology

"Sealing Technology. Abandoned exploration and monitoring drill holes extending into the disposal formation, together with the injection well, constitute possible pathways to the biosphere. These penetrations of the containment formation are particularly critical for the deep well liquid injection concept where the waste is in a pressurized mobile form. Until techniques are proven for sealing drill holes, care must be exercised in drilling candidate areas. In particular, all drill holes must be accurately surveyed, both at the surface and by down-hole instruments, and records of drillings, casing losses, casing strata, hole closure and backfilling must be maintained.

is likewise inconsistent with the extensive discussion of this problem in Section 1.3 where the waste is not in a pressurized mobile form. (154)

Response

Section 6.1 of the final Statement has been significantly revised and improved with the purpose of facilitating comparative assessment between alternative concepts and to establish consistency within the Statement. However, because of the relatively lower state of development of the eight alternatives to that of the mined repository, equal depth of treatment

ALTERNATIVE DISPOSAL CONCEPTS

is not possible. Section 6.2 of the final Statement addresses the issue of technical uncertainty in its presentation of comparative assessment of alternatives.

Draft p. 3.8.24

Issue

Has reverse well injection in salt beds been considered? Salt is not mentioned among media under consideration. It has been found to be amenable to bedding plane fracture. (218-DOI)

Response

Reverse well injection in salt has not been considered since the methods generally involve use of water which is normally considered incompatible with a salt host medium to be used for waste disposal.

Draft p. 3.8.25

Issue

Under Data Needs the statement

"Data Needs. Present geological knowledge is adequate for generic studies and identification of a number of candidate sites. Site-specific studies will require more detailed investigations, in particular the stratigraphy and lithology of the disposal and containing formation, geochemistry of strata and fluids, and structural features."

is not consistent with similar discussions in Section 3.1. (154)

Response

Section 6.1.6.3 of the final Statement discusses the status of technical development and R&D needs for the well injection concept. Under the heading Development of Criteria for and Categorization of Siting Opportunities present geologic knowledge is judged suitable only for preconceptual generic studies and identification of candidate site (emphasis added).

Draft p. 3.8.29

Issue

In view of grout injection experience at ORNL and liquid chemical injection at other U. S. Government locations, the statement made in paragraph 4 concerning the lack of adequate monitoring data is not appropriate. (124)

ALTERNATIVE DISPOSAL CONCEPTS

Response

In view of grout injection experience at ORNL and liquid chemical injection experience at other U. S. Government locations, some monitoring data are available. However, the lack of data completeness and lack of consensus on data interpretation at these well sites supports the Statement's position.

Issue

Reverse-well Disposal involves the injection of solutions or slurries without the great expense of verification and encapsulation. A shortcoming of the concept is that the boundaries of the repository are determined by the local characteristics of the shale and will not be precisely predictable. If, however, the solution or slurry were pumped into a mined cavern that had been carefully inspected and tested this shortcoming would be avoided. If the reprocessing plant were constructed directly above the repository, the risks of waste handling and transport would be minimized. (6)

Response

The use of a slurry in mined cavern would be a variation on the waste form from which would be used in mined disposal. This waste form would probably not be compatible with the multibarrier concept for the conservative design of waste package systems. For a discussion of the requirements for one waste form and waste package, see Section 5.1.2 of the final Statement.

Transmutation (a)

Draft Section 1.3.9 and 3.9

Issue

Section 1.3.9 (and Section 3.9) describes an option for partitioning and transmutation. A variant of this, which could be utilized near term would be pre-solidification treatment and separate packaging of the more highly radioactive species (e.g., Cs, Sr, Te, etc.). This should be discussed in the GEIS, since it does not require the development of transmutation technology, and would facilitate the recovery of the many useful resources in the waste. (198)

(a) This concept was referred to as "Partitioning and Transmutation" in the draft Statement. The discussion of partitioning as a predisposal treatment alternative can be found in final Section 4.3.2.

ALTERNATIVE DISPOSAL CONCEPTSResponse

Utilization of separated constituents from spent fuel waste has been studied for many years and no practical demand has developed. There appears to be no urgent current reason to separate unsalable fractions. The concept of partitioning highly radioactive species can be applied in conjunction with the space disposal option, as discussed in Section 6.1.8.

Draft Section 3.9.1Issue

Several commenters noted that in a discussion of the disadvantages of the transmutation option, the Statement did not make a clear presentation of the pros and cons of the concept. (58, 124)

Response

In the preparation of the final Statement, efforts were made to provide a balanced discussion of the advantages and disadvantages of the concept using the most current information available in the technical literature.

Draft Section 3.9.1Issue

In Section 3.9.1.2 the disadvantages of increased thermal loading is misleading. In fact, it may be advantageous to separate the high heat producing isotopes (e.g., dilution to control volumetric heat generation of such nuclides). It is suggested that the counter argument to this disadvantage be included in the list of advantages. (124)

Response

The statement was misunderstood here. Partitioning certain high level isotopes reduces the bulk waste heat generation rate. Partitioning of long-lived transuranics reduces the time at which the bulk wastes are radiologically significant. Disadvantages, as pointed out in the Statement, center on the added costs and risks to both occupational groups and general population.

Draft Section 3.9.2Issue

It is difficult to recognize the future importance of transmutation. Since the authors of the EIS have apparently utilized availability as an important parameter for applicability in disposing of nuclear waste, it is a foregone conclusion that transmutation cannot be taken seriously. (124)

ALTERNATIVE DISPOSAL CONCEPTSResponse

This factor is an important reason for the low potential indicated in Section 6.2 of the final Statement. A number of other apparent deficiencies are also inherent to the concept as presented in Section 6.1.7.

Draft Section 3.9.2Issue

This very well written section throughout puts partitioning and transmutation into reasonable perspective. It clearly notes in several places (for example at page 3.9.1) that "partitioning may be a pre-disposal option, but it can never be a final disposal option by itself."

In this sense the discussion is out of place in the same way that Section 3.2 is. Neither is a disposal option and they should not be presented in the Final EIS on the same level as alternatives which might become disposal options.

The only quarrels we have at all with this section as written are:

- 1) It should be relegated to a Predisposal Option Appendix.
- 2) It should state the obvious conclusion which comes through at nearly every page--this predisposal option has no usefulness in the near (30 years) term and its likelihood of every being cost-beneficial is very close to zero. (154)

Response

The comments are well taken. Transmutation as treated in the revised Section 6.1.7, however, does have some potential as a disposal option in the elimination of the long-lived actinides but has a number of drawbacks. Practical achievement of this potential appears questionable. The discussion of partitioning is now in Section 4.3.2.1 on predisposal options of the final Statement.

In the revised version of the GEIS, Section 6.2, Comparative Assessment of Disposal Alternatives, relegates Transmutation to a very low order of priority as an alternative for ultimate waste disposal. This evaluation and its presentation essentially confirm the comment.

Draft Section 3.9.2Issue

Partitioning and Transmutation - not applicable because of transportation costs, increased cost of waste treatment, and it does not eliminate the need for final disposal of the waste. (88, 121)

ALTERNATIVE DISPOSAL CONCEPTSResponse

The factors enumerated certainly would need to be considered in evaluating partitioning and transmutation. If the long-lived actinides could be fully eliminated in a reasonable time period, the approach might be worth the cost. It does not appear, however, that we can achieve that goal.

Draft Section 3.9.2Issue

The last sentence in the first paragraph should be changed to read "may require." The certainty of the statement depends on potential uses of transmuted wastes. Future requirements may not have been adequately analyzed at this time. (124)

Response

The certainty of the original statement still seems appropriate, since the residual product of actinide transmutation remains a mixture of fission products. Although some of these may find uses in the future, the bulk of them will have to be isolated securely from the environment.

Draft Section 3.9.3Issue

It is difficult to verify that decommissioning and disposal of hazardous reagents create additional complicated radionuclide logistics. Keeping programs in perspective, decommissioning impacts for reference fuel cycles are highly speculative, and thus to discuss nuclear waste streams as being a disadvantage is highly speculative (from a decommissioning standpoint) and should be omitted. (124)

Response

It is agreed that little practical experience exists for decommissioning of fuel cycle facilities. However, the intent of the statement was to point out, at least indirectly, that an effective transmutation program requires a considerable number of reactor recycles to completely transmute the candidate radionuclides, since only 5 to 7 percent are transmuted in a typical cycle. Thus the number of fuel cycle facilities which must ultimately be decommissioned is considered larger than other concepts. Also since the chemical processes used for separation (partition) of nuclides prior to re-cycle are complex, an increased volume of contaminated reagents, relative to most other processes will be generated. Similarly, different waste systems are introduced by the processing complexity.

ALTERNATIVE DISPOSAL CONCEPTS

Therefore, the conclusion drawn was that the number of facilities, increased volumes and greater diversity of waste streams would imply a more complicated waste management program.

Draft Section 3.9.5

Issue

The length of this section (3.9.5) could be reduced to that of the last paragraph since most of the remaining information has been previously discussed. (124)

Response

A revised version of the section on partitioning and transmutation appears as Section 6.1.7, Transmutation. This revision should be found more succinct. In particular, secondary impacts from partitioning are reflected in an expanded section, 6.1.7.4, Impacts of Construction and Operation (Preemplacement).

Draft p. 3.9.9

Issue

The discussion of iodine partitioning is somewhat misleading. Partitioning is accomplished by air-sparging of the dissolver solution to force the last few percent of iodine into the off-gas. The Iodex process is one of several possible methods for recovery of iodine from the off-gas. The hazard posed by the 1% or less that is not removed from the solution should be quantified and put in perspective.

The consideration of the disposal of recovered iodine appears to have failed to include consideration of isotopic dilution as an added protection against hazards arising from possible release from geologic confinement at some time in the "distant future." Emplacement of iodine wastes, either in seabed sediments or salt depositories (assuming iodine content similar to seawater), would provide a high degree of protection against the possibility of significant future up-take of ^{129}I by any individual. (58)

Response

The discussion of iodine partitioning now appears in Section 4.3.4.2, Gaseous Radionuclide Recovery. The new Transmutation Section 6.1.7 deals with actinide partition and transmutation. Accordingly, there is no specific treatment of the ^{128}I hazard, but the general treatment in 4.7.2.3, Radiological Effects of Reprocessing Fuel Cycle Waste Management, is considered adequate to the purpose of the GEIS. The points made pertaining to mitigation of the potential hazard of iodine release are certainly valid.

ALTERNATIVE DISPOSAL CONCEPTSDraft p. 3.9.11Issue

It appears that the concern expressed over escape of ^{14}C from a geologic repository at some time in the "distant future" is more qualitative than quantitative. The basis for such concern should be qualified in relationship to natural ^{14}C background, including consideration of half-life, isotopic dilution, and biological uptake. It would appear that disposal options affording isotopic dilution (e.g., seabed sediments) would warrant more serious consideration than space disposal. (58)

Response

The consequences of C^{14} release are treated in Section 4.7.2.3 of the final Statement, Radiological Effects of Reprocessing Fuel Cycle Waste Management. Since the population exposure appears relatively minimal, a more detailed treatment does not appear warranted.

Draft p. 3.9.19Issue

Since the actinide recycle gives higher neutron activities, it would appear to create more problems than it solves. (35)

Response

It was pointed out in the Statement that the enhanced neutron activity in recycled actinides does exist and could be a problem. Whether it creates more problems than it solves would need to be analyzed in greater depth.

Space DisposalDraft p. 1.30Issue

Leaving the waste in a solar orbit or disposing of it on the moon still leaves the waste available for future exposure to man. Would not direct injection into the sun be preferable? (218-DOI)

Response

Although it would be technically feasible to inject the waste into the Sun, this is not possible with present launch vehicles. Alternatively, indirect flight to the Sun from the

ALTERNATIVE DISPOSAL CONCEPTS

Earth could be accomplished with present vehicles by using planetary swing-by trajectories. However, this approach is considered not practical because of limited opportunities to launch from the Earth.

It is highly unlikely (see Section 6.1.8 of the final Statement) that waste successfully placed in a solar orbit situated halfway between that of the Earth and Venus and inclined to the plane of the ecliptic would pose significant hazards to the Earth's environment and to human health and safety.

Draft p. 3.10.1-48

Issue

This well written section makes it clear that this option cannot be used soon enough to be a candidate for initial disposal and the Final EIS should so state. It is unfortunate that the candid admission from page 3.9.9 "However, there is no firm technical basis at this time for asserting that the disposal of any or all of these elements in geologic form (Section 3.1) is either inadequate or undesirable" was not included in this section also. (154)

Response

The commenter's points are well taken. The reader is referred to Section 6.2 of the final Statement for a comparative assessment of the attributes of the alternative concepts and the mined repository.

Draft p. 3.10.1

Issue

Ohio has previously indicated its dissatisfaction with this aspect of the report. It should be added that experimental data should exist on plutonium dispersion from space due to the SNAP reactor breakdown in the early sixties, and, from upper atmospheric bomb testing data from about the same period. (35)

Response

Specific experimental or empirical information collected from atmospheric dispersal of radioactive materials following nuclear weapons tests; or following the accidental reentry and burnup of a plutonium heat source in the 1960's; or following the recent reentry of the Russian Cosmos satellite, was not used directly in the preparation of Section 6.1.8 of the final Statement. The reader is referred, however, to Section 6.1.8.4 for a discussion of the impacts of accidents, including accidental reentry and launch pad accidents, that might cause atmospheric dispersal of radioactive materials.

ALTERNATIVE DISPOSAL CONCEPTSDraft p. 3.10.1Issue

The waste mix information provided in Figure 3.10.1 is confusing; (a) Zircaloy should not be included in Mix 2; (b) it is not clear if Zircaloy hulls are mixed with HLW subsequent to reprocessing. (124)

Response

It is agreed that the presentation of waste mix information might be confusing. Zircaloy hulls should not be considered as a part of the waste that would be disposed of in space. These hulls may contain a significant fraction of high-level and transuranic wastes which would be disposed of through use of another alternative concept (e.g. the mined repository). Alternatively, new processing technology would be needed to strip the high-level and transuranic waste from the Zircaloy so that these wastes could be reasonably disposed of using the space disposal concept.

Draft p. 3.10.2Issue

The last sentence in Section 3.10.1.1 should be modified to include cost considerations, if applicable. If cost was not considered, then the sentence should indicate that the conclusions refer to technical considerations only. (124)

Response

Cost considerations for the space disposal concept are discussed in Section 6.1.8.6, Cost Analysis, in the final Statement. Section 6.2 of the final Statement considers cost attributes in the comparative assessment of the alternate concepts and the mined repository.

Draft p. 3.10.9Issue

Based on basic experience with the space program, it seems possible and highly desirable to discuss the cost and natural resource utilization needed for this alternative along with other information for the upper stage operations presented on Page 3.10.9 (124)

Response

Cost and resource utilization information has been included in Section 6.1.8 of the final Statement. Due to the limited current state of development of the space disposal concept, this information applies to the overall concept and is not broken down for detailed elements of that concept.

ALTERNATIVE DISPOSAL CONCEPTSDraft p. 3.10.18Issue

Space disposal - too costly - \$22 million per launch - the danger of launch pad and reentry accidents. (88, 121)

Response

The commenter makes reference to two important considerations for the selection of an acceptable alternative for the disposal of radioactive waste. For a comparative analysis of the attributes, including costs and disadvantages, of alternative concepts and the mined repository concept, the reader is referred to Section 6.2 of the final Statement.

Draft p. 3.10.23Issue

The last sentence in the second paragraph of Section 3.10.2.6 (Page 3.10.23) should be modified to indicate that, based on the disposal methods and from the information presented in the EIS, space disposal cannot at this time be justified from an economic or risk standpoint. (124)

Response

Section 6.1.8 of the final Statement presents information that reflects the current state of knowledge regarding the space disposal concept. Section 6.2 of the final Statement presents an assessment of the alternative concepts, including space disposal, using a comparison of attributes and related performance standards. Both Section 6.1.8 and Section 6.2 address the issues of costs and risks in providing comparative information regarding the concepts considered.

Draft p. 3.10.23Issue

The third paragraph under 3.10.2.6 should also include in the total integrated system risk, nonradiological risks associated with space facility construction, as well as all manufacturing and support phases required for the additional launchings dictated by the space disposal option. It appears highly doubtful that the incremental gains afforded by the isolation of space disposal can outweigh the total risks involved in achieving that isolation. (58)

ALTERNATIVE DISPOSAL CONCEPTSResponse

The revised space disposal section of the final Statement discusses both facility construction environmental impacts and routine operation nonradiological impacts in Section 6.1.8.4. This section does not specifically mention all manufacturing and support phases required for the additional launchings dictated by the space disposal option, but it does conclude that environmental impacts from normal operations or nonradiological accidents from a space disposal option are not expected to be significant.

Draft p. 3.10.26

Issue

The fourth paragraph should be expanded to indicate that all necessary precautions, including conservative engineering design, will be implemented to ensure that the probability of an accident will be small and the consequences less. (The last statement in that paragraph can be interpreted as meaning that calculations will be performed to show that the probability is vanishingly small.) (124)

Response

Several sections of the final Statement address various aspects of the accident issue. In Section 6.1.8.3 the final Statement states that the major technology requirements are in design for safety. This section also points out that initial R&D should include an assessment of unique safety and environmental aspects of the space disposal concept (e.g., launch pad fires and explosions affecting the waste package). Section 6.1.8.4 is concerned with impacts of construction and operation (preplacement), including risks unique to this concept and the types of reliability data needed for risk assessments.

Draft p. 3.10.26

Issue

Note this understatement of the section: Moreover, legal concerns may lengthen the amount of time needed to actually execute a space disposal option. (154)

Response

Many, likely formidable, legal and institutional issues would require resolution before this concept could be implemented. Currently available information is, however, insufficient to ascertain the complexity of these issues. The statements of concern to the commenter remain in the final Statement to ensure that this important area is addressed. In addition, Section 6.2 of the final Statement presents a comparative assessment of the alternative concepts and the mined repository that compare attributes, including legal and institutional factors, of these concepts.

ALTERNATIVE DISPOSAL CONCEPTSDraft p. 3.10.27Issue

The socioeconomic aspects discussed on the top of p. 3.10.27 should be quantified. Most other EIS documents have included these types of data. (124)

Response

The Socioeconomic Impacts discussion in Section 6.1.8.4 of the final Statement notes that a detailed assessment of these impacts would require more accurate employment estimates, information on the industrial sectors affected by capital expenditures, and identification of the specific geographic areas involved; and, that these types of detailed information would not be available unless the definition of this concept were further advanced. This discussion does note, however, that the current estimate of launch rates suggest that the support of the entire space transportation system would require 25,000 to 75,000 employees. The range of these estimates reflects the present level of definition of this concept.

Draft p. 3.10.28Issue

It is recommended that the data in Table 3.10.5 be modified with less precise percentage results. Per the supporting documentation, it seems that presentation with such precision is unjustified. (124)

Response

This information is presented with more general estimates in the final Statement, Section 6.1.8.4.

Draft p. 3.10.30Issue

Processes for the production of calcine waste have indeed been demonstrated on an engineering scale with radioactive material for many years at Idaho Chemical Reprocessing Plant. (154)

Response

Although processes for production of certain waste forms (e.g. calcine) have been demonstrated on an engineering scale as stated by the commenter, the selection of specific

ALTERNATIVE DISPOSAL CONCEPTS

waste components and incorporation of those components into a suitable waste form for space disposal has not been done. For additional discussion of R&D needs for this topic see Section 6.1.8.3.

Draft p. 3. 10.30Issue

The added R&D required for glass forms identified in the ninth paragraphs should be rationalized and quantified. One would be forced to conclude that the concerns over glass stability must be related to post-abort status. If this is the case, it would appear that quantification of the overall risk would either show aborts to be of overriding concern or the potential instability problems of glass to be insignificant. (58)

Response

The revised presentation, Section 6.1.8.4, identifies high altitude reentry and burnup with potential dispersion of the payload as particles in the atmosphere as an important accident, to be considered with regard to radiation exposure to the general public. These potential conditions would establish requirements for waste form characteristics, and possible waste form R&D. R&D requirements are discussed in Section 6.1.8.3.

Draft p. 3.10.37Issue

The U and/or Pu disposal method defined should be clarified (Section 3.10.5.2). Per information presented on p. 3.10.4, these isotopes are disposed in space. (124)

Response

Section 6.1.8 of the final Statement provides discussion of the wastes that are best suited for disposal in space. Clarification is made of the possible use of space disposal as a component of a total waste management system. Such a system would necessarily include another disposal alternative (e.g. mined geologic repositories) for those wastes which, for a number of reasons, would likely not be suitable for disposal in space.

Draft p. 3.10.46Issue

The R&D requirements seem quite modest compared to estimates for other alternatives. Surely much of the potential cost is not stated. (154)

ALTERNATIVE DISPOSAL CONCEPTSResponse

The full cost of R&D that would be required to implement the space disposal concept has not been determined. Section 6.1.8.3 of the final Statement presents a summary of the efforts thought to be necessary to provide a basis for such a cost estimate. The efforts required to provide the basis are estimated to cost \$20 million.

Draft p. 3.10.46Issue

Per review of Table 3.10.6, this disposal mode is not being seriously considered as a viable alternative to terrestrial disposal. If it is to be considered as a likely alternative, additional funds would be necessary for engineering design and safety assessments. (124)

Response

Section 6.1.8.3 of the final Statement presents a discussion of the status of technical development and R&D needs for the space disposal concept. Noteworthy is the fact that current information is considered insufficient to support a meaningful estimate of the full cost of R&D that would be required to implement this concept.

Draft Appendix RIssue

This is an interesting discussion but the state of development of the technology does not permit more than qualitative information. (113-EPA)

Response

Although the current state of development of the space disposal concept is considered to be insufficient to provide quantitative estimates of many important performance characteristics, there are data available in the literature to provide limited quantification of impacts and some costs. This information is presented in Section 6.1.8 of the final Statement.

Issue

The National Aeronautics and Space Administration (NASA) submitted many comments on the space disposal concept in the draft Statement. In general, the comments suggested rephrasing or revising the text to better represent the current status of the development of the space disposal concept. (200)

ALTERNATIVE DISPOSAL CONCEPTSResponse

The comments were taken into consideration in preparing the final Statement, Section 6.1.8.

HEARING BOARD REPORT AND RESPONSES

This section presents the Hearing Board Report and responses to issues raised in this report. The Hearing Board Report represents oral testimony presented by interested persons, organizations and agencies through a series of public hearings held to air comments on the Draft Environmental Impact Statement of April 1979 "Management of Commercially Generated Radioactive Waste." As a result of this testimony the Hearing Board identified issues that should be addressed during preparation of the final Statement.

For the convenience of the reader, this section is structured such that the responses are shown opposite the issues raised in the report.

HEARING BOARD CHARTER

The principal responsibility of the Hearing Board was to conduct a series of public hearings giving interested persons, organizations, and agencies the opportunity to comment on the Department of Energy (DOE) Draft Environmental Impact Statement of April 1979, "Management of Commercially Generated Radioactive Waste." Based on public testimony briefings, and supplementary documents, the Board was to prepare a report to DOE identifying issues that should be addressed during preparation of the final Environmental Impact Statement.

Other than the Chairman, the Hearing Board was composed of distinguished scientists. None of the hearing board members were employees of DOE; they were contracted as consultants to conduct public hearings and identify to DOE those issues requiring consideration or revision during preparation of the Final Environmental Impact Statement. Furthermore, none of the hearing board members were involved in the preparation of the Environmental Impact Statement and were thus disinterested, impartial reviewers.

HEARING BOARD MEMBERS

Professor George T. Frampton served as the Hearing Board Chairman for the public hearings. Professor Frampton, a lawyer, specializes in the field of atomic energy law and has taught and performed research in this area for many years. He has also written articles dealing with this specialized area of law. Professor Frampton's past experience in the nuclear field includes serving on the Illinois Atomic Power Investigating Commission, and being a consultant to the U.S. Joint Congressional Committee on Atomic Energy, to the National Committee on Radiation Protection and Measurements (regarding legislative control of ionizing radiation), and to the Atomic Energy Commission and its successor agency, the Energy Research and Development Administration.

Another board member, Dr. Dorothy K. Newman, is a sociologist, a socioeconomist, and a distinguished author. She received her Ph.D in Sociology from Yale University. She was director of the Energy in People's Lives study, for the Energy Policy Project of the Ford Foundation. She has been a socioeconomist with the U.S. Department of Labor in matters relating to construction of housing, consumer economics and issues of the disadvantaged. She has been Director of Research for the National Urban League; Director of the Project on Race and Social Policy, supported by the Carnegie Corporation; and she recently completed a book entitled "Protest, Politics and Prosperity, Black Americans and White Institutions, 1940 to 1975." She is presently a consultant and lecturer.

Another board member, Dr. Clifford V. Smith, is Vice-President for the Administration at Oregon State University and Professor of Environmental Engineering. He received his Bachelor of Science in Civil Engineering from the University of Iowa, his Masters of Science in Environmental Engineering and his Ph.D in Radiological Science and Environmental Engineering at Johns Hopkins. He is a Registered Professional Engineer. He has 26 years of experience in all facets of environmental engineering with industry, government and universities. He was formerly Director of the Office of Nuclear Materials, Safety and

Safeguards of the Nuclear Regulatory Commission and a member of the interagency Review Group, which in 1979 completed an extensive study of nuclear waste management for the President of the United States. Dr. Smith's past experience includes being a Regional Administrator of the Environmental Protection Agency in the Pacific Region, Deputy Regional Administrator in New England, and he has been associated with environmental issues related to nuclear power going back to the Shippingport Nuclear Power Plant in 1959.

Dr. Hubert L. Barnes, another board member, is the author of about 70 technical papers and books on the geochemistry of mineral resources, including a book on "Uranium Prospecting." He has been a Guggenheim Fellow at the Geochemistry Institute of the University of Goettingen, Exchange Scientist of the U.S. National Academy of Sciences to the Russian Academy of Sciences, and Chairman of the United States National Committee for Geochemistry. He is a Fellow of the Geological Society of America and of the Mineralogical Society of America. He is also a Professor of Geochemistry, and Director of the Ore Deposits Research Institute at The Pennsylvania State University. Past experience includes geochemical research on the origin of hydrothermal ore deposits at the Geophysical Laboratory of the Carnegie Institute of Washington, and extensive consulting on geochemical problems in environmental pollution and in the search for mineral resources. He also is the inventor of patented methods of air pollution control and of crystal growth. He has held several distinguished lectureships at universities in the United States and in Europe and now with the Society of Economic Geologists.

Another hearing board member was Dr. Melvin W. Carter who received his Bachelor's and Master's of Science Degree at Georgia Institute of Technology, and his Ph.D at the University of Florida. He is currently at the Georgia Institute of Technology as Director of the Center for Radiological Protection and Professor at the School of Nuclear Engineering. He has had more than 29 years of experience in the radiation protection field as an environmental engineer and health physicist. He worked for the United States Environmental Protection Agency. He has been assigned major responsibilities including Director of the Southeastern Radiological Health Laboratory, and the Director of the National Environmental Research Center, Las Vegas. In addition to numerous articles and books, he has been co-author of a monograph and also editor of Environmental International. He has served as a consultant to federal and state agencies, as well as major industrial concerns and international organizations, and belongs to a number of professional and honorary organizations, including the Health Physics Society, the National Council on Radiation Protection and Measurements, and the International Radiation Protection Association.

HEARINGS AND TESTIMONY

The Board conducted hearings in five major cities: Washington, DC (June 26-27, 1979); Chicago, Illinois (August 8-9, 1979); Atlanta, Georgia (September 25-26 1979); Dallas, Texas (October 2-3, 1979); and San Francisco, California (October 8-9, 1979). Notice of public availability of the Statement appeared in the Federal Register of April 20, 1979. Later

notices in the Federal Register (June 1, July 3, and July 18, 1979) provided hearing dates and places, invited written comments, and gave other information about public participation.

A total of 142 witnesses testified before the Board. In addition, the Board reviewed more than 200 written comments, some of which were extensive. These were submitted instead of or as an extension of oral testimony. Most of the presentations were by individuals expressing their personal views. Many respondents had technical, scientific or medical expertise. Many represented concerned citizens groups, state or local governments, nuclear research and service organizations, utilities, and trade associations. Several federal agencies submitted extensive written reviews. Transcripts of the hearings are available for public inspection at certain local and regional offices of the Department of Energy and at its Headquarters in Washington, DC.

Based on public testimony, review of supplemental documents and briefings, the following report was prepared.

HEARING BOARD REPORT
ON THE DEPARTMENT OF ENERGY
DRAFT ENVIRONMENTAL IMPACT STATEMENT

entitled

"MANAGEMENT OF COMMERCIALY GENERATED RADIOACTIVE WASTE"
(DOE/EIS-0046-D)

CONTENTS

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February 1980

	Page
I. BACKGROUND: THE HEARING BOARD AND PROCEDURES	1
II. RECOMMENDATIONS	5
A. PURPOSE OF THE STATEMENT	5
B. SCOPE	6
1. Relationship of the Statement to Other Nuclear Activities	6
2. Treatment of Alternatives	7
3. Sequence of Comparative Environmental Analyses by Stages, Beginning with Spent Fuel	9
4. Testing and Experimental Programs	9
5. Humanistic Concerns and Consequences	10
6. Other Nuclear Power Projections	12
C. STATEMENT PRESENTATION	12
1. Length	12
2. Summary	13
3. Conclusions and Recommendations	13
4. Editorial and Terminological	14
D. SPECIFIC SUBJECTS	15
1. Risk-Benefit Analysis	15
2. Reprocessing	17
3. Schedules and Timing	17
4. Costs	18
5. Types and Characteristics of Host Rock	19
E. PUBLIC NOTIFICATION PROCEDURES	21
F. MISCELLANEOUS ITEMS REQUIRING REVIEW	23
III. CONCLUSIONS	25
IV. BACKGROUND REFERENCES: REPORTS PROVIDED TO THE BOARD BY THE U.S. DEPARTMENT OF ENERGY	26

HEARING BOARD REPORT
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DEPARTMENT OF ENERGY
DRAFT ENVIRONMENTAL IMPACT STATEMENT
entitled

"MANAGEMENT OF COMMERCIALY GENERATED RADIOACTIVE WASTE"
(DOE/EIS-0046-D)

I. BACKGROUND: THE HEARING BOARD AND PROCEDURES

The Board's main responsibilities were to conduct a series of public hearings to give interested persons, organizations, and governmental agencies an opportunity to comment on the Department of Energy Draft Environmental Impact Statement of April 1979, "Management of Commercially Generated Radioactive Waste", (called the "Statement" in this Report), and to prepare a report to the Department identifying the significant issues that should be addressed in preparing a final statement.

The Department of Energy selected the members of the Hearing Board with the intention of obtaining the public's views through an outside, impartial, diverse group with substantive knowledge and experience about energy and the environment. Professor George T. Frampton, Sr., Professor of Law and formerly a Vice-Chancellor of the University of Illinois, chaired the Board. Other members were Dr. Hubert L. Barnes, Professor of Geochemistry and Director of the Ore Deposits Research Institute of the Pennsylvania State University; Dr. Melvin W. Carter, Professor of Nuclear Engineering and Director of the Center for Radiological Protection at the Georgia Institute of Technology; Dr. Dorothy K. Newman, socio-economist and author of studies about the

energy consumer and other citizen concerns for the U. S. Department of Labor, for the National Urban League as former research director, and for Carnegie Corporation of New York as Director of the Project on Race and Social Policy; and Dr. Clifford V. Smith, Professor of Environmental Engineering and Vice President, Oregon State University, and former Director of the Office of Nuclear Materials, Safety, and Safeguards of the Nuclear Regulatory Commission.

The Board conducted hearings in five major cities: Washington, D. C. (June 26-27, 1979); Chicago, Illinois (August 8-9, 1979); Atlanta, Georgia (September 25-26, 1979); Dallas, Texas (October 2-3, 1979); and San Francisco, California (October 8-9, 1979). Notice of public availability of the Statement appeared in the "Federal Register" of April 20, 1979. Later notices in the "Federal Register" (June 1, July 3, and July 18, 1979) provided hearing dates and places, invited written comments, and gave other information about public participation. These issues of the "Federal Register" are cited in the Background References at the end of this report.

The regional offices of the Department of Energy advertised the subject of the hearing and the time and location in news media of each region before the first day of each hearing. The Department also made extensive mailings to various public interest groups and organizations in an effort to elicit their views.

A total of 142 witnesses testified before the Board. In addition, the Board reviewed more than 200 written comments, some of them extensive. These were submitted instead or in extension of oral testimony. Most of the presentations were by individuals expressing their personal views. Many respondents had technical, scientific or medical expertise. Many represented concerned citizens' groups, state or local governments, nuclear research and service organizations, power companies, and trade associations. Several Federal agencies submitted extensive written reviews. Transcripts of the hearings are available for public inspection at certain local and regional offices of the Department of Energy and at its Headquarters in Washington, D. C.

The Board members attended a briefing in June 1979 by the Department of Energy Division of Waste Isolation and by Pacific Northwest Laboratories of the Battelle Memorial Institute, which prepared the draft statement for the Department of Energy. In further preparation, Board members read the Statement (Volume 1) "Management of Commercially Generated Radioactive Waste" (about 700 pages) and its supporting "Appendices" (Volume 2), about 650 pages. Further backup volumes to the draft statement were submitted later for the Board's review. These include five volumes on "Technology for Commercial Radioactive Waste Management" (DOE/ET-0028) and three volumes on "Environmental Aspects of Commercial Radioactive Waste Management" (DOE/ET-0029). The ten volumes total more than 5,000 pages.

Associated documents that came to the Board's attention were "Nontechnical Issues in Waste Management: Ethical, Institutional, and Political Concerns" (PNL-2400) and "Safety Indices and Their Application to Nuclear Waste Management Safety Assessment" (PNL-2727). The Board also reviewed other documents relevant to the draft environmental impact statement, including the "Report to the President by the Interagency Review Group on Nuclear Waste Management" (March 1979, TID-29442). Dr. Smith, a member of this Hearing Board, served on the Interagency Review Group.

Other reports submitted to the Board members for their information were the report by the Comptroller General of the United States on "The Nation's Nuclear Waste--Proposals for Organization and Siting" (June 21, 1979, EMD-79-77), the two-volume "Draft Environmental Impact Statement, Waste Isolation Pilot Plant" (April 1979, DOE/EIS-0026-D), and the Waste Isolation Pilot Plant Hearing Board report of November 6, 1979, on its series of public hearings. Other environmental impact statements came to the Board's attention because they illuminated parts of the waste management problem. These are about power reactor spent fuel, including the storage of foreign power reactor spent fuel, and the program plan for defense waste management. These and other documents (cited in the Background References at the end of this report) are within the scope of the Department of Energy's total "Nuclear Waste Management Program" (April 1979, DOE/ET-0094). Particularly important for assessing the place of the environmental impact statement about the management of commercially generated nuclear waste is the "Commercial Waste Management Multi-Year Program Plan" (August 1979).

II. RECOMMENDATIONS

The Board observes that the Department and its contractors have prepared a statement of substantial depth and breadth, made efforts at its wide distribution, and solicited public participation in deliberations about the Statement. However, on the basis of the public testimony and briefings, and supplementary documents, the Board identifies the following issues for development or modification in the final generic environmental impact statement.

A. PURPOSE OF THE STATEMENT

The Board recommends that the Department of Energy define the purpose of the Statement more clearly at the outset to avoid the obvious confusion reflected in the oral testimony and written comments.

The purpose of the "Draft Environmental Impact Statement" is unclear at the outset and requires clarification. The Foreword states: "This Generic Environmental Impact Statement (GEIS) is intended to provide environmental input" for the "decision" "of selecting an appropriate programmatic strategy leading to the permanent isolation of commercial radioactive wastes in a fashion that provides reasonable assurance of safe, permanent isolation of the material." The term "generic" (and its effect on the entire document) is not defined or explained, but it becomes part of the acronym, "GEIS", adopted by the Department of Energy.

Response

Chapter 2.0 (Introduction) of this document has been modified to more clearly identify this Statement's purpose and need. This chapter also addresses the "generic" nature of the document with revisions made to explain the use of this term.

Further, this sole statement of purpose confused some people by covering, but failing to distinguish between, two important purposes of an environmental impact statement. One purpose is to demonstrate that the environmental consequences of a proposed federal action have been considered by identifying and describing them and comparing them with the consequences of alternative courses of action. The other purpose is to subject that demonstration to public review and comment, thus affording broad participation in a decision before action is taken. While the statement of purpose of the document does not, on its face, cover the second purpose, use of the word "input", and the process of distributing the documents and holding hearings, are evidence of the Department of Energy's intention to encompass both purposes.

B. SCOPE

1. RELATIONSHIP OF THE STATEMENT TO OTHER NUCLEAR ACTIVITIES.

The Statement should reveal early how its limited scope relates to other waste management operations and to the processes of nuclear technology here and abroad.

The title of the Statement does not limit the discussion to the sole problem of finding a strategy for permanent and safe isolation of commercially generated radioactive wastes. The broad term "management" was included in the title "Management of Commercially Generated Radioactive Waste", and the term "high-level" was excluded. Many witnesses and readers, therefore, expected to find information about the total system, all wastes, and explicitly, various techniques for disposal and their environmental impacts.

Response

Section 2.2 (Relationship to other Waste Management Decisions) was added to this Statement's Introduction to inform the readers in a general way about the total waste management system and how this environmental impact statement relates to this system.

2. TREATMENT OF ALTERNATIVES

Only the most viable alternative strategies should be assessed in detail. Less viable alternatives should be ranked for feasibility, and the fullness of their treatment should be commensurate with that ranking.

The Statement identifies ten alternative strategies for waste isolation. Two--chemical resynthesis and transmutation--are not techniques for waste isolation, but are methods of waste treatment. They are not, therefore, alternative courses of action. Furthermore, even if transmutation were a method of waste treatment, it is not technologically or economically feasible in the foreseeable future (twenty years) and thus could have been treated less comprehensively on that basis alone.

Of the remaining eight options, only those with some reasonable prospect of relatively short-term economic and technical feasibility should require fully detailed analysis. Less detailed treatment of other disposal options, while not ignoring any positive findings of current research, should indicate the critical factors that reduce feasibility in this century. None of the alternatives is entirely without merit, especially in changed economic or political circumstances. At present, however, near-term feasibility is a vital consideration but is in doubt for certain options. Examples are: for space disposal, risk and costs; for ice sheet disposal, international jurisdictional and treaty obstacles; for seabed disposal, current legal restrictions, (e.g. the Marine Sanctuary Act of 1972), and transportation risk; for island disposal, geologic instability and transportation risk; for deep hole disposal, costs; for rock melting, potential release of volatiles; and for well injection, seismic risk. Furthermore, nonretrievability is an additional concern to all the above except island disposal.

Response

The section on the Comparison of Disposal Technologies was revised so that the reader can more easily distinguish between disposal technologies. Restructuring the section also aids the reader in determining the feasibility of disposal options and identifies which options should receive further program emphasis. Part of this restructuring consisted of placing "Chemical Resynthesis" and "Partitioning" in Section 4.3 (Waste Treatment and Packaging).

Not even those in the scientific community and in federal or state agencies, and few among the general public, knew about the Department of Energy's comprehensive waste management plan and schedule, the numerous radioactive waste management environmental impact statements already written or in progress, and the extent of experimental work under way. It is no wonder that countless hours at the hearings were spent on issues that were not pertinent to the isolation of commercially generated high-level waste. If the Statement at the outset had informed readers in a general way, as introduction, about the total management system, considerable confusion would have been avoided. For example, readers need an early explanation of the difference between low-level and high-level wastes and how and where they are handled and stored, the programs for handling defense wastes and their relationship to commercial waste management, experimental work under way in all spheres, issues surrounding waste arriving from abroad, what happens to nuclear waste generated by medical and research activities, what happens to non-nuclear waste from commercial nuclear facilities, and what happens upon the decommissioning of a nuclear facility. After such an exposition the focus of the Statement about commercially generated high-level radioactive waste would be in context, as would the concern--frequently expressed--that testing or experimental work on nuclear waste was insufficient or lacking.

Response

The Introduction (Chapter 2.0) presented in the final Statement was revised to provide the reader with a general overview of the total waste management system and identifies those areas of this total system which are addressed by this Statement. Discussion of the various waste types and methods for handling and storage of these wastes are presented in Section 4.1. A perspective of the impacts from handling defense wastes in relation to those from commercial waste management is presented in the Summary of the final Statement.

3. SEQUENCE OF COMPARATIVE ENVIRONMENTAL ANALYSES BY STAGES,
BEGINNING WITH SPENT FUEL

The entire chain of environmental consequences of managing commercially generated high-level wastes should be treated consistently and compared for each viable alternative course of action, beginning with spent fuel at the reactor.

The Statement does not provide information at each appropriate stage of the entire spent fuel cycle about environmental consequences, beginning with on-site storage and going through chemical treatment, encapsulation, handling, transportation, site selection, testing, emplacement, and storage.

4. TESTING AND EXPERIMENTAL PROGRAMS

Since the feasibility of proposed options must be verified by appropriate experiments, the Statement should disclose more fully and explain the nature and extent of the testing and experimentation now under way and planned.

Experimental work already under way in this country and abroad, including the extent of U.S.-international cooperation, should be more adequately treated in the final Statement. In addition to the work in Sweden mentioned in the Statement, experiments are also under way in India, West Germany, France, Great Britain, Canada, Japan, and the Soviet Union, among others.

Response

The draft Statement did analyze the entire cycle of post-fission waste management activities. However, to ensure that the reader understands the scope of the analysis carried out, the final document is structured such that the predisposal activities (waste treatment and packaging, waste storage, waste transportation, and decommissioning) are presented first (see Chapter 4.0) and disposal activities are then outlined in subsequent chapters (Chapters 5.0 and 6.0).

Response

Ongoing and planned research and development efforts in this country and abroad have been addressed in a revised section on Technical Feasibility (Section 5.2). Where deemed appropriate, experimental work in other countries has been mentioned in this Statement.

An international exchange of information is provided through a cooperative program with the United States, Canada, Sweden and West Germany as the participants. Ongoing and planned research and development efforts in this country and abroad have been addressed in a revised section on Technical Feasibility (Section 5.2) of the final Statement.

A common misconception is that holes drilled into a potential repository would destroy its integrity permanently. Therefore, discussion is necessary of how such holes are effectively sealed by cementing or grouting and how permeability tests by injecting water at high pressure are used to assess the final repository conditions.

Abandoned mines, which provide a wide variety of rock types as possible repository hosts, with presumably limited environmental consequences, have been suggested by the U. S. Bureau of Mines for testing, and are used for that purpose in Sweden and West Germany. Information about the present use or consideration of such mines should be provided.

5. HUMANISTIC CONCERNS AND CONSEQUENCES

Ecological, social, psycho-social, political, and economic consequences should be given more prominence and receive more professional attention.

The significance of social concerns and their political influence is apparent in the testimony of witnesses ranging from the pro-nuclear to the anti-nuclear. Witnesses emphasized, and the Board concurs, that the degree to which human concerns are taken into account could result in the success or failure of any waste management plan.

Response

The structure for Volume 1 was revised to draw information relevant to nontechnical issues relating to waste management in general from several places in the draft together into one section (Section 3.5). In addition, the Technology Comparisons section (Section 6.2) uses domestic policy considerations and international policy conflicts as criteria upon which the disposal options were examined and evaluated. Nontechnical issues relevant to the site selection process are covered briefly in Section 5.1.

The Statement, however, deals inadequately with the humanistic effects of each stage in the process of each alternative management system, especially with those stages before waste reaches a final site. The Statement deals with social and economic consequences of constructing and operating a storage facility more fully, but even in this instance the treatment is sketchy. For instance, insufficient attention is given to the interaction with waste management operations of alternative employment and unemployment projections nationally and regionally, of migration streams, and of easily demonstrable demographic changes, to mention a few conditions considered too summarily or not at all. Neglected for each option are occupational opportunities, training requirements, and hazards involved in handling, shipping, encapsulating, and inserting and recovering wastes.

Even in addressing issues surrounding only site selection and operation, detail is lacking on how participation by State and local governments and the public takes place in the experimental or final site selection. The Statement is sketchy on environmental surveillance, monitoring, and managing each kind of site. The problems to be encountered in clean-up in the event of decommissioning or serious accident, as well as possible evacuation, require more detailed analysis taking into account comparative environmental effects before and after the event, for each option, with emphasis on behavioral and biological science approaches.

Response

In considering social impacts, unemployment and employment projections, migration streams and demographic changes were taken into account in estimating the demographic impact of various waste management and disposal facilities. Great effort was made in selecting different reference environments so that these factors could be considered. The impacts of decommissioning waste management facilities were also analyzed to ensure greater clarity. All of these discussions were placed in a single section (Section 5.4).

Specific plans relating to institutional arrangements for control of disposal sites were not offered because such plans are site specific and no site has been selected. However, Section 3.5 offers discussions of general institutional arrangements which could apply to any site.

In summary, humanistic considerations and consequences require much more sophisticated development and more social imagination.

6. OTHER NUCLEAR POWER PROJECTIONS

The Statement should take account of the environmental effects of managing waste generated by other projections of nuclear power production.

Additional projections beyond those now considered and which require attention are (a) that all commercially generated nuclear power production would cease in 1980; and (b) that nuclear power production by present facilities or those currently licensed would be permitted only until their normal decommissioning dates.

C. STATEMENT PRESENTATION

1. LENGTH

The Statement is unnecessarily wordy and voluminous. It should be reorganized and cut drastically by judicious rewriting and editing, with the aim of reducing the basic volume to less than 300 pages.

Much of the Statement treats methods in inordinate detail, thus obscuring the findings and the central ideas that went into models. It relies

Response

The Statement quantitatively analyzed the waste management impacts of the following five scenarios:

1. Present inventory (equivalent to industry shutdown).
2. Present capacity to retirement (equivalent to licensing no new reactors).
3. Installed capacity of 250 GWe in year 2000 and declining to zero in year 2040.
4. Installed capacity of 250 GWe in year 2000 and continuing at 250 GWe to year 2040.
5. Installed capacity of 250 GWe in year 2000 and growing to 500 GWe in year 2040.

These scenarios bound the range of possible reactor futures that are presently thought to be reasonable.

Response

An effort was made to increase the overall clarity and readability of the document by reducing the number of pages, being consistent in the use of terminology, utilizing summary tables where possible, and relegating supporting or more detailed data and information to the Appendices.

heavily on jargon, acronyms, convoluted sentences and bulky tables and figures, much of which could be dispensed with by distillation into simple English. Many parts of the Statement are redundant, confusing, and sometimes conflicting. Reorganization could follow in part the guidelines for environmental impact statements of the Council on Environmental Quality.

2. SUMMARY

After revising the Statement, a short, clear, concise, accurate and readable summary should be prepared that is comprehensive and reflects the findings of the Statement as a whole. An even shorter summary should also be prepared aimed at fuller comprehension by the general public.

Many who testified or wrote comments had read only the Statement Summary from Volume 1. It appeared to the Board that few read Volume 1 and fewer Volume 2. Almost none had seen the other eight volumes. It is important that a short summary carry the essential message clearly.

3. CONCLUSIONS AND RECOMMENDATIONS

Conclusions or recommendations should be recognizable as such, without equivocation or hedging.

The conclusions and recommendations should be presented in a positive and straightforward manner, thus assisting the reader in determining what is important, what is known, the degree of that knowledge, and the major thrusts of the Statement.

Response

In preparation of the final Statement, the Department of Energy (DOE) has taken several steps to be responsive to the recommendations on the organization and presentation of the draft Statement. The structure of Volume 1 was modified to focus on the proposed Federal action and to make more evident the systems aspects of the Statement.

This was accomplished by:

1. Outlining the purpose and need of the Statement (Chapter 2). This chapter discusses the intent of the document, the proposed Federal action, and the "decision territory covered."
2. Identifying programmatic alternatives (Chapter 3) including a statement of a no-action alternative which the draft Statement did not do.
3. Developing Chapter 4, which discusses predisposal options and systems.
4. Emphasizing the proposed action (mined geologic repository) by discussing in a separate Chapter (5) with the presentation of disposal alternatives in Chapter 6.
5. Including a new Chapter 7, which discussed tradeoffs between the proposed action and the two alternative actions on a complete system basis.

The Summary chapter was extensively revised in order that the material would be more easily understood by the general public. In revising this chapter DOE was sensitive to the comments that the significant conclusions be highlighted in the Summary and that they be substantiated by the material in the text. An effort was also made to increase the overall clarity and readability of the document by reducing the page length, being consistent in the use of terminology, utilizing summary tables whenever possible, and relegating supporting data or information to the appendices.

4. EDITORIAL AND TERMINOLOGICAL

The Board recommends:

- Minimizing the use of acronyms and defining them when first used and in the glossary.
- A simple page-numbering system without competing section numbers.
- A clear, well-organized table of contents.
- A comprehensive index.
- A complete glossary with readable definitions.
- Identifying the key persons involved in preparing the Statement.
- Distinguishing clearly between "containment" and "isolation."
- Not using the title "Conventional Geologic Disposal" to denote a disposal not yet "conventional". More accurate would be "Disposal in Mined Repositories".
- Using "Well Injection" instead of "Reverse Well Disposal"
- Not calling spent fuel "waste", since spent fuel has intrinsic energy value.
- Changing the phrase "Geologic Emplacement Following Chemical Synthesis" to "Waste Solidification".
- Distinguishing between individual radiation dose equivalents and accumulative dose to populations. An amount of man-rem per individual, for example, is a contradiction in terms.

Response

Editorial and terminological recommendations made by the Board were considered during preparation of the final Statement and the appropriate changes were made.

D. SPECIFIC SUBJECTS

1. RISK-BENEFIT ANALYSIS

The Statement should acknowledge and define the special problems of undertaking risk-benefit analysis in this unusual area, pursue the analysis in an orderly fashion, and recognize and include nontechnical values, by integrating political and social concerns with technical consideration.

Frequent criticisms and misperceptions communicated to the Board involved the analysis of risk. Many thought risks unduly minimized; many thought the opposite. Such a range of views results from fragmentation of the analysis of risk in the Statement, an overly simplistic technical approach, and lack of sufficient accommodation to nontechnical considerations.

Traditional assessment of environmental consequences attempts to analyze hazards and risks quantitatively in relation to benefits. The limitations and difficulties of such quantification in this area, however, require special caution and consideration which do not appear to have been brought to bear in preparing the Statement. A wider range of risks should be assessed, including some ordinary, realistic situations as well as some least expected. Some worst case accidents used for analysis (the meteorite, for example) are too extreme.

Response

Section 3.4 of the final Statement provides the reader with some perspectives on the concepts of "risk" as it pertains to radioactive waste management. If proper comparisons are made, the reader is given a perspective of the relative impacts of radiological wastes versus the hazards of other materials in the environment. Such comparisons are not intended to justify the potential impacts from radioactive waste management.

This Statement presents radiological impacts of unintended events first from a consequence viewpoint. The frequency of occurrence of the particular event is then identified so that if one desires to determine the expected impact from such an event one may do so. However, the consequences of an event (in the absence of the frequency of occurrence) are an upper-bound estimate of the impacts. The document also identified other criteria in Section 6.2 (in addition to radiological impacts) that the "decision-maker" would consider in evaluating the various disposal options.

For the disposal technologies analyzed, this Statement presents radiological impacts under the sections titled "Environmental Analysis of Construction and Operation" (5.4, 6.1.1.4, 6.1.2.4 . . . 6.1.8.4) and "Environmental Analysis Over Long Term" (5.5, 6.1.1.5, 6.1.2.5 . . . 6.1.8.5). Radiological impacts of predisposal activities are addressed in Sections 4.7 and 4.8. Chapter 7.0 outlines radiological impact for entire waste management systems.

425

Several views of comparative risk need airing and reconciliation. One view suggests that risks from commercially generated radioactive waste and its management activities are additive (cumulative) so that comparison with other risks masks a true danger. This view requires consideration.

At the other extreme, but necessary, is more complete comparison of radiation exposure from high-level radioactive waste with risk from other conditions or materials. It should include more information about the waste exposure with that from the original ore, natural background at varying elevations, and with other minerals or poisons (as with pesticides, nitrates from fertilizers, sulphur dioxide from flue gases, and the like). At this level of analysis, comparative dosage rates and a definitive basis for evaluation are critical.

In another dimension is the desirability of comparing the risks from the radioactive waste management system with the risks from other waste-generating systems, such as coal, metal mining, or logging, for examples.

Assessment of total risk for each alternative waste-isolation option is necessary and is lacking. More imaginative concepts or analogies are required when specific data are not available. In the case of marine transport for island disposal, automobile shipment could be used as an analogy for marine transport of cask-sized units. Comparison would then be possible with continental disposal, for which transport is over land near sizeable populations, whether by rail or by truck.

2. REPROCESSING

The final Statement should summarize in one place the comparative waste management implications both of continuing and of discontinuing the present moratorium on reprocessing.

The consequences of the present moratorium against reprocessing of spent fuel elements are not explicitly summarized. A change in this reprocessing policy, viewed by many as inevitable, would make portions of the Statement obsolete. The costs and benefits, risks, and time associated with planned retrievability should be included in the Statement.

The high-level wastes associated with possible breeder reactors should also be described in comparison with those from other sources.

Response

Chapter 7.0 of the final Statement examines waste management impacts from a systems viewpoint. System impacts are outlined for various nuclear growth scenarios, repository availability dates, and commercial fuel cycles (i.e., once-through--no reprocessing, U and Pu recycle--reprocessing).

Retrievability is designed for increasing assurance of program safety, not for the purpose of recovery for future reprocessing. The only materials which are to be placed in a repository are those which have been fully characterized as waste. A determination as to whether spent fuel is a waste product or a resource will be made prior to a repository becoming operational.

Response

Various spent fuel storage methods and associated storage periods are examined in Section 4.4 of the final Statement. The initial storage interval permits decay of short-lived radionuclides which results in a lower heat generation component. This Statement examines initial storage periods from 20 years up to 100 years. However, the time period for which spent fuel will require storage prior to disposal ultimately depends on issues such as environmental considerations, storage capacity and repository availability.

The DOE Position Paper to the NRC rulemaking hearings on nuclear waste storage and disposal^(a) places the first repository availability date between the years 1997 and 2006 depending on the outcome of future decisions.

Section 3.4.1 of the final Statement examines risk and risk perspectives. For this issue "nominal" risk is assumed to be the risk represented by the original uranium ore used to produce the fuel. Spent fuel's relative toxicity index decreases by 550-fold in the first 500 years from reactor discharge and then decreases 20-fold in over 999,000 years remaining to one-million years out of reactor. The toxicity of spent fuel can be compared to that of the original uranium ore that was mined to produce the fuel. A typical ore contains 0.2% U_3O_8 ; some 3400 tons (or 1200 m³) is mined to produce one ton of fuel. The toxicity of Spent fuel is 14-fold above this at 1000 years. The HLW toxicity equals the toxicity of the ore that produced it at 1500 years.

The text has been thoroughly revised with attention paid to such consistency. The "500 year" and "700 year" numbers are from the referenced work of other authors. The value used in this Statement is 1000 years.

(a) Department of Energy Statement of Position to the NRC Rulemaking on Nuclear Waste Storage and Disposal, April 1980.

3. SCHEDULES AND TIMING

The Statement should make consistent and clear the estimated time during or at which certain events will occur.

How long will spent fuel be stored before disposal? How long does it take for radioactive waste to become a "nominal" risk? There should be a consistent use of half-lives or time intervals (not a variation among 500, 600, 1,000 and 10,000 years, for example) when evaluating hazards from radioactivity.

The Statement does not clarify the time frame in which a failure or inability to implement a "permanent" solution to the waste isolation problem will begin to alter the environmental consequences of the present "temporary" isolation by on-site storage. The uncertainty on this matter left witnesses free to speculate that a long-continued resort to on-site storage would have effects ranging from none in the next few years to a foreseeable exhaustion of on-site facilities and a consequent shutdown of nuclear power production.

4. COSTS

The cost analysis in the Statement should be more comprehensive, and should relate to the whole system of each alternative so as to provide a basis for cost comparisons.

Costs should be more fully analyzed to take into account the entire system for every viable isolation alternative. They should include administration, research and development, interim storage, encasement and the cost of encasement materials, vehicles, transport, training, labor, negotiations leading to site selection, risks and risk insurance, land, construction, final emplacement, institutional surveillance, and emergency preparedness.

Response

The basis for the analysis of costs is explained in Section 3.2.8 of this final Statement. The estimates which are intended to represent the ultimate cost to the consumer of electric power are the systems costs given in Section 7.0 of the Statement and summarized in the executive summary. These costs are intended to be as comprehensive as is possible in keeping with the generic nature of the statement. For the purpose of clarity, the final Statement presents predisposal cost estimates in Section 4.9 and repository cost estimates in Section 5.6. Cost estimates for the alternative concepts are presented in Sections 6.1.1.6 through 6.1.8.6. Included in the predisposal and final disposal costs estimates are land acquisition, construction and decommissioning, labor, encasement and other materials, utilities, vehicles and other transportation costs, administration and other overhead, insurance, taxes (where applicable), permits, licenses, financing, and allowances for contingencies. Costs of any continuing repository surveillance that may be required are insignificant in relation to the rest of the waste management costs developed here. Complete waste management system costs are presented in Section 7.6.

Since current costs in developing and producing a feasible waste isolation program could be perceived as a nuclear power subsidy, comparative analysis with other regulated systems and large-scale enterprises could be made as, for example, railroads, airlines, and automobiles.

Costs are affected by the availability of materials here and abroad. In the array of materials proposed as canisters, for example, there may be problems of cost and access to necessary quantities of such metals as titanium, zirconium, gold, platinum, nickel and others.

5. TYPES AND CHARACTERISTICS OF HOST ROCK

The final statement should be modified to reflect that an objective evaluation has been made for all potential host rocks.

More present information about one alternative over another does not necessarily translate into a clear-cut technological advantage nor support a preference for, or emphasis on, any one rock type. The present Statement, moreover, reflects an emphasis on salt which may not be supported even by all the facts currently available.

Types

The advantages and disadvantages of all host rocks as repositories, including salt, should be compiled and compared objectively.

Response

This final Statement has been revised to reflect more clearly the objective evaluation of candidate disposal media. It is pointed out that salt has had relatively more investigation than other candidate media at this time, but as with the others, it is only recommended for further investigation. This Statement considers that repositories are potentially feasible in all media. However, medium selected based on its physical properties will have to be in a favorable and safe geological environment.

The discussion of the four rock types presented in this Statement is not intended to exclude any geologic media as a repository host rock, but rather to be representative of the potential geologic disposal media. Tuff will therefore receive consideration as a potential disposal medium. Examination of the four rock types (salt, shale, granite and basalt) provide the foundation for determining the suitability of geologic media for nuclear waste storage and disposal as well as providing a focus for a research and development program for geologic disposal.

Factors that might be considered in evaluating salt include: thermal conductance, fluid migration toward warm canisters, high acidity when hot, high plasticity, low sorptive capacity, susceptibility to radiation damage, permeability evidenced by braccia pipes, and environmental problems associated with surface storage or disposal of salt.

A fifth rock type, tuff, should be added to the other four--granite, basalt, shale and salt--as a possible candidate host rock for repositories. If anhydrite is included, its large volume change on hydration should be considered.

Mobility

The mobility of wastes in various rock types deserves greater emphasis in the Statement. Should a canister be breached, the viability of the host rock as a backup barrier directly depends on this mobility. Solubility under reducing conditions is likely to be dominant in limiting the long-term migration of components from a breached canister.

Sorption

Since the sorption characteristics and reactivity of host rocks to radioactive solutes are among the most important properties of the

multiple barrier concept, they should be more clearly developed. In this connection, shales could be superior to other proposed rock types provided that there is not large-scale lateral migration of ground water through the shale.

Permeability

Large-scale permeability is of concern for all rock types. Thus, Table 3.1.1 should be revised to show only bulk properties. Shale across bedding is much less permeable than typical broken basalt flows or ash beds.

E. PUBLIC NOTIFICATION PROCEDURES

When the U. S. Department of Energy holds further hearings, it should consider major modifications in the public notification procedures used for these hearings.

Although the Board recognizes that substantial effort was made to circulate the Statement and obtain public comment on it, the Board nevertheless recommends critical review of the Department's entire notification and mailing procedures. Its mailing lists should reach a more diversified group. Advertising, if used at all, should be more effectively designed and placed. Copies of impact statements should be available farther in advance of hearing dates.

Response

Recommendations by the Hearing Board to modify public notification procedures to reach a more diversified group will be considered by the DOE and where applicable will be implemented.

While the hearings on the Statement were under way, the Board was surprised to learn from witnesses that hearings were taking place at a nearby location on the environmental impact statement on the Waste Isolation Pilot Plant (WIPP), a proposed facility which would be used to test some of the uncertainties related to this Statement. Both the timing and the locations of those hearings competed for the attention of interested persons.

There seems also to be lack of consistency and consideration in time given for thoughtful public reaction to the Statement as compared with its preparation. For example, evolution of the Statement has taken several years during which a number of related waste management program documents have been prepared, received, and in one case, withdrawn. In contrast to the time taken in those processes, the limited period--days or weeks--given to interested persons to prepare comments in writing or orally on this massive technical document has imposed an unnecessary condition of undue haste on public participation.

These procedural deficiencies offset the earnestness with which many individuals in the Department of Energy sought wider public participation and designed a hearing process for doing so. The deficiencies also created the unfortunate impression at times that effective public participation is not regarded as a serious part of revising the Statement or of decision-making in the nuclear waste management program.

F. MISCELLANEOUS ITEMS REQUIRING REVIEW

1. Maps.

Maps of the United States showing the distribution of salt deposits and granites omit areas of their respective rock types and should either be deleted or more carefully compiled. Because several types of metamorphic rocks could behave toward waste in a similar manner to granites, perhaps they might also be identified on the map of granites. The map should clarify whether it refers only to granites or also to granodiorites and similar rocks, since granites are rarely homogeneous in composition. The map of basalts also omits many areas of basalt in the west and the Appalachians, for example.

2. Chemical Resynthesis.

The indication in the Statement that chemical resynthesis is for the purpose of achieving equilibrium with the host rock needs revision. The foreign compounds of the waste cannot be in true equilibrium with the host rocks in the thermodynamic sense. Actually, the goal of resynthesis is to achieve minimum kinetic mobility of waste components in the host.

3. Retrievability from Wells.

After well injection, widespread dispersal of the waste fluid down the hydrologic gradient occurs, so that no significant fraction of the fluid is likely to be recovered by pumping. Furthermore, the reaction of other components of the host rock to neutralize any acid flush would thwart efforts to leach radioactive solutes lost by reaction with the host rock.

Response - Item 1

Maps showing the distribution of salt, granite and basalt have been revised to show extensive deposits of these rock types in the United States and are intended to give the reader a feeling for the spacial orientation of these rock types in this country. Additional revisions were made, where deemed necessary, to clarify the material being shown by the maps.

Response - Item 2

The Department agrees and has revised portions of this section to more accurately explain the goal of chemical resynthesis.

Response - Item 3

This Statement points out under the discussion of well injection, that utilization of this alternative would preclude waste retrievability.

4. Erosion Rates.

For most of the United States, erosion rates are only a few centimeters per thousand years, so that, contrary to the discussion of erosion rates in the Statement, erosion is no threat to mined or similarly deep repositories.

5. Geothermal Gradients.

The effect of geothermal gradients on the emplacement depth of waste canisters should be addressed because of the effects on heat loss from canisters and on rock plasticity.

6. Canister Rupturing.

Canister integrity can be affected by circumstances other than chemical corrosion or tectonic events. Some of such circumstances are puncturing and mechanical stress caused by differential compaction of the host rock on the canister overpacking.

7. Risk from Rock Falls.

The statement that "accidents that threaten human life are rarely caused by failure of the rock itself" does not square with the fact that rock falls, not rock bursts, are the typical causes of mining fatalities. These falls occur generally at joints or faults that have caused local weaknesses in the host rock.

Response - Item 4

It is generally agreed that erosion is not likely to be a threat to a deep repository. Much of the Statement is deliberately conservative to overly conservative in approach, particularly the accident scenarios.

Response - Item 5

The geothermal gradient has not received detailed consideration in this Statement. The contribution to the total heat of the system is a small fraction of that generated by the waste and would therefore not be a significant consideration when assessing the effects of heat loss from canisters and on rock plasticity.

Response - Item 6

Canister rupturing can occur in ways other than those discussed in this Statement. However, the Department feels that the mechanisms identified and described in this Statement adequately cover the range from potential mechanical to chemical failure of the canister.

Response - Item 7

Agree. The text has been changed to reflect that rock falls are the typical cause of mining fatalities.

III. CONCLUSIONS

Subject to the above recommendations, suggestions, and comments, the Hearing Board concludes that:

1. The Statement seriously and impressively analyzes the environmental impacts of proposed actions for solving the problem of disposing of commercially generated high-level radioactive waste.
2. The Statement has served effectively as a vehicle for public comment and for indicating and generating changes that should be made in the final statement.
3. The Statement supports the conclusion, in principle, that commercially generated high-level radioactive waste can be disposed of by one or more alternative strategies with minimal and acceptable environmental consequences, and that the present preferred disposal option is a deep, mined geologic repository.

The Board is concerned that the longer the delay in implementing an appropriate strategy to solve the problem of high-level radioactive waste disposal, the greater the erosion of political, scientific, and public support essential for such a strategy.

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APPENDIX A

LIST OF RESPONDENTS TO DOE/EIS-0046D

APPENDIX A

LIST OF RESPONDENTS TO DOE/EIS-0046D

Appendix A contains the names of individuals, organizations and agencies that submitted written comments on the draft Statement. A number has been assigned to each letter. These numbers appear following each issue stated in the text of Volume 3, as a means of indicating the respondents concerned with that issue.

To assist the reader, two lists are presented. The first list is ordered by letter number which correspond to the sequence in which the letters were recieved by the Department. The second list is an alphabetized list of the names, organizations and agencies which responded to the draft Statement with written comments. This second list is divided such that private individuals and organizations appear first, followed by State and Federal Agencies. This list was alphabetized by name for individuals and organizations responding, by agency for Federal respondents and by State and agency for State respondents.

<u>COMMENTER</u>	<u>DATE RECEIVED</u>
1. Mr. R. C. Baxter Allied-General Nuclear Services P.O. Box 847 Barnwell, SC 29812	05-31-79
2. Ms. Edith Roth 6029 Oakdale Avenue Woodland Hills, CA 91367	06-06-79
3. Mr. Terry Kubicek Natural Resources Coordinator State Office of Planning & Programming Box 94601, State Capital, Room 1321 Lincoln, NB 68509	06-27-79
4. Mr. Arne O. Kauranen Regional Engineer Federal Energy Regulatory Commission Regional Office 730 Peachtree Street, N.E. Atlanta, GA 30308	06-27-79
5. Mr. W. J. Mecham Argonne National Laboratory 9700 South Cass Avenue Argonne, IL 60439	06-27-79
6. Mr. W. P. Bebbington 905 Whitney Drive Aiken, SC 29801	07-03-79
7. Mr. William P. Dornsife, P.E. Bureau of Radiation Protection Department of Environmental Resources P.O. Box 2063 Harrisburg, PA 17120	07-03-79
8. Mr. William A. Lochstet 119 E. Aaron Dr. State College, PA 16801	07-06-79
9. Chrys Baggett, Director State Clearing House North Carolina Department of Administration 116 W. Jones Street Raleigh, NC 27611	07-06-79
10. Mr. Donald Orth Savannah River Laboratory P.O. Box A Aiken, SC 29801	07-10-79
11. Mr. Ted Breitmayer 148 Draeger Drive Moraga, CA 94556	07-10-79

<u>COMMENTS</u>	<u>DATE RECEIVED</u>
12. Ms. Edith M. McKee 416 Maple Street Winnetka, IL 60093	07-10-79
13. Mr. W. S. Geiger 741 Woodhill Drive Lakeland, FL 33808	07-10-79
14. Ms. Pam Demo-Rybus Division of Budget, Policy Planning and Coordination Executive Office of the Governor State Clearinghouse State House Boise, ID 83720	07-10-79
15. Mr. Paul Kiepe 2141 1st Avenue S. Payette, ID 83661	07-11-79
16. Mr. Max Eisenberg, Acting Director Environmental Health Administration Department of Health and Mental Hygiene P.O. Box 13387 201 West Preston Street Baltimore, MD 21203	07-11-79
17. Mr. Arvind Srivastava Kaiser Engineers Inc. 300 Lakeside Drive Oakland, CA 94666	07-11-79
18. Mr. Edward Gabriel Council of Energy Resource Tribes One Thousand Connecticut Avenue, N.W. Suite 610 Washington, DC 20036	07-11-79
19. Ms. Emily Neary State of Vermont Office of the Governor State A-95 Clearinghouse Montpelier, VT 05602	07-11-79
20. Mr. T. J. Crossley 16 Loma Linda Lane Lakeland, FL 33803	07-11-79
21. Mr. Robert K. Kunita 4234 Lake Ridge Drive Raleigh, NC 27604	07-11-79
22. Mr. Donald E. Harley Office of the Governor Executive Office Building 411 West 13th Street Austin, TX 78701	07-11-79

COMMENTER	DATE RECEIVED
23. Mr. Sidney R. Galler United States Department of Commerce The Assistant Secretary for Science and Technology Washington, DC 20230	07-11-79
24. Mr. Mike Nolan Governor's Office of Planning Coordination Capitol Complex Carson City, NV 89710	07-12-79
25. Mr. Paul W. Levy Brookhaven National Laboratory Associated Universities, Inc. Upton, NY 11973	07-12-79
26. Mr. Robert H. Moen 6602 Tam O'Shanter Drive San Jose, CA 95120	07-12-79
27. Mr. Daniel Hunt National Science Foundation Washington, DC 20550	07-12-79
28. Mr. Derek Wallentinsen Albuquerque Group Sierra Club 3131 Quincy NE Albuquerque, NM 87110	07-13-79
29. Mr. John E. Schulte 1916 Marconi Road Wall, NJ 07719	07-13-79
30. Mr. Marvin I. Lewis 6504 Bradford Terrace Philadelphia, PA 19149	07-13-79
31. Mr. Paul DeGaeta Department of Administration Division of State Planning and Research Fourth Floor--Mills Building 109 W. 9th Topeka, KS 66612	07-16-79
32. Mr. Lawrence Schmidt, Chief Office of Environmental Review Department of Environmental Protection P.O. Box 1390 Trenton, NJ 08625	07-16-79
33. Mr. B. Jim Porter Office of Conservation P.O. Box 14690 Baton Rouge, LA 70808	07-16-79

<u>COMMENTS</u>	<u>DATE RECEIVED</u>
34. Mr. D. L. Renberger Assistant Director of Technology Washington Public Power Supply System P.O. Box 968 3000 George Washington Way Richland, WA 99352	07-16-79
35. Ms. Judith Y. Brachman Administering Officer State Clearinghouse 30 East Broad Street 39th Floor Columbus, OH 43215	07-17-79
36. Chrys Baggett, Director (Duplicate of 9) State Clearinghouse North Carolina Department of Administration 116 West Jones Street Raleigh, NC 27611	07-18-79
37. Mr. J. B. Jackson, Jr. Administrator Commonwealth of Virginia 903 Ninth Street Office Building Richmond, VA 23219	07-19-79
38. Mr. J. A. McBride Box 192 Merrifield, VA 22116	07-23-79
39. Mr. Thomas M. Webb, Director Natural Resources Planning Tennessee State Planning Office 660 Capitol Hill Building 301 Seventh Avenue, North Nashville, TN 37219	07-23-79
40. Mr. Joel Yellin Harvard University John F. Kennedy School of Government 79 Boylston Street Cambridge, MA 02138	07-25-79
41. Dick Hartman State Planning Coordinator Wyoming Executive Department Cheyenne, WY 82001	07-27-79
42. J. S. Sherman Department of Environmental Regulation Twin Towers Office Building 2600 Blair Stone Road Tallahassee, FL 32301	07-31-79
43. Lee Sherman Dreyfus Governor, State of Wisconsin State Capitol Madison, WI 53702	08-06-79

<u>COMMENTER</u>	<u>DATE RECEIVED</u>
44. Joseph M. Sorboro NEFCO 137 South Main Street Suite 300 Delaware Building Akron, OH 44308	07-29-79
45. James Mulloy Department of Water and Power 111 North Hope Street Los Angeles, CA 90012	08-08-79
46. Donald E. Harley, Manager (Addresses wrong EIS) Economic and Natural Resources Budget and Planning Office Office of the Governor Executive Office Building 411 West 13th Street Austin, TX 78701	08-08-79
47. Mrs. Lawrence Lewis 3206 Columbia Parkway Cincinnati, OH 45206	08-09-79
48. Harold V. Larson (Addresses wrong EIS) Battelle, Pacific Northwest Laboratories P.O. Box 999 Richland, WA 99352	08-14-79
49. J. H. Kittel Argonne National Laboratory 9700 South Cass Avenue Argonne, IL 60439	08-14-79
50. Mr. Gerald R. Day, Executive Director Illinois Commission on Atomic Energy Lincoln Tower Plaza 524 South Second Street--Room 415 Springfield, IL 62706	08-15-79
51. Mrs. Lawrence Lewis (Additional comments to 47) 3206 Columbia Parkway Cincinnati, OH 45206	08-17-79
52. Albert H. Quie Governor of Minnesota State of Minnesota St. Paul, MN 55155	08-20-79
53. Charles L. Pater Belden Corporation 2000 South Cass Avenue Geneva, IL 60134	08-21-79

<u>COMMENTER</u>	<u>DATE RECEIVED</u>
54. Edward Hennelly Savannah River Laboratory Aiken, SC 29801	08-30-79
55. Catherine Quigg, Research Director Pollution & Environmental Problems, Inc. Box 309, Palatine, IL 60067	09-05-79
56. John Dilday 200 Hillsboro St. Cary, NC 27511	09-07-79
57. Ermont M. Lawrence 5241 S.E. 115th Street Belleview, FL 32620	09-10-79
58. Lyle L. Zahn, Jr., Manager Spent Fuel Services Operation General Electric Company 175 Curtner Avenue San Jose, CA 95125	09-10-79
59. J. R. Buford 237 Leland Way Hanford, CA 93230	09-12-79
60. Paulette Oakes 1118 Ingra Anchorage, AK 99501	09-12-79
61. Carl Groff 200 Bolinas RD #81 Fairfax, CA 94930	09-12-79
62. Brandt Mannchen 4055 South Braeswood #303 Houston, TX 77025	09-12-79
63. Louis B. Carrick 365 Beechwood Drive Athens, GA 30606	09-18-79
64. Walter E. Wallis Wallis Engineering 1954--R Old Middlefield Way Mountain View, CA 94043	09-18-79
65. Agnes S. Easterly 1100 University Street Seattle, WA 98101	09-19-79
66. Robert F. Davis 2706 N. Chamberlain Chattanooga, TN 37406	09-19-79

<u>COMMENTS</u>	<u>DATE RECEIVED</u>
67. Roland Livernois Livernois Enterprise 1231 Kenyon Road Lake City, GA 30252	09-19-79
68. Ron Fahs, Director (Duplicate of 42) Intergovernmental Coordination State of Florida Office of the Governor Tallahassee, FL 32301	09-19-79
69. L. E. Lantaff, I.I.C U.S. Department of Agriculture Food Safety and Quality Service Meat and Poultry Inspection Program 1400 S. Green Street, P.O. Box 744 Henderson, KY 42420	09-20-79
70. Herbert L. Harper Executive Director and State Historic Preservation Officer Tennessee Department of Conservation Tennessee Historical Commission 4721 Trousdale Drive Nashville, TN 37220	09-20-79
71. David K. Brooks P.O. Box 16929 Jackson, MS 39206	09-20-79
72. Mr. Ben Cornwell 1795 Magnolia Tree Germantown, TN 38138	09-20-79
73. David DeBus, PhD. 425 Bonair Street La Jolla, CA 92037	09-20-79
74. Linda Baziuk Route 3, Box 438 Yadkinville, NC 27055	09-20-79
75. David Shields 419 Oakdale Rd. N.E. Atlanta, GA 30307	09-19-79
76. Arden A. Richards 3305 East Skylane Drive Florence, SC 29501	09-20-79
77. Edward L. Jones Route 4, Berry Road Taylors, SC 29687	09-20-79
78. Bob Brister 4308 Rhodes Avenue Memphis, TN 38111	09-20-79

COMMENTER	DATE RECEIVED
79. Walter Julius Weems 3599 Saturn Drive Northwest Atlanta, GA 30331	09-20-79
80. Paul Grier 10460 S.W. 111 Street Miami, FL 33176	09-21-79
81. Mr. J. P. Moore 6107 Main Avenue Tampa, FL 33611	09-21-79
82. Kenneth Yonovitz 4206 Washington Road West Palm Beach, FL 33405	09-21-79
83. Marjorie King Carolina Nutrition Corner, Inc. 2514-A. E. North Street, Ext. Burns Corner (E. North St. & Pelham Rd) Greenville, SC 29607	09-21-79
84. Leo Benson, III 1274 Sledge Avenue Memphis, TN 38104	09-21-79
85. Harry A. Galliher 250 West Artesia Pomona, CA 91768	09-24-79
86. William J. McClusky 4317 Rhodes Memphis, TN 38111	09-24-79
87. Daniel Kaminsky Savannah River Galleries 309 Bull Street Savannah, GA 31401	09-24-79
88. Zelia Jensen, R.N. Route 1, Box 86A Grandview, TN 37337	09-25-79
89. Chris Findlay 235 Stone Avenue Lexington, KY 40508	09-25-79
90. Lorane H. Minis 4142 Amsterdam Circle Savannah, GA 31405	09-25-79
91. Mrs. Marge Nejedly 109 W. Kennedy Drive Holiday, FL 33590	09-26-79

<u>COMMENTS</u>	<u>DATE RECEIVED</u>
92. Fred L. Lemon Highway 90 and Church Street Ocean Springs, MS 39564	09-26-79
93. Beth E. Buzzard 2016 E. State Avenue Olympia, WA 98506	09-26-79
94. Dr. Leo Pocus, M.D. The Doctors Clinic 2512 Wheaton Way Bremerton, WA 98310	09-26-79
95. Charles W. Larimer, Jr. 1308 60th St. Ensley Birmingham, AL 35228	09-26-79
96. Nadine A. Vesel 6237 N. Winthrop Chicago, IL 60660	09-26-79
97. George Busbee Office of the Governor Atlanta, GA 30334	09-26-79
98. Gloria C. Rains Manasota--88 5314 Bay State Rd. Palmetto, FL 33561	09-26-79
99. Mrs. Irma Ragan Holland 2210 Creston Road Raleigh, NC 27608	09-26-79
100. Calvin P. Ragan Route 1--Box 323 New Hill, NC 27562	09-26-79
101. Yvonne Brunton 120 Leatnice Lane Vicksburg, MS 39170	09-26-79
102. Elizabeth Briggs 10 Bliss Avenue Woodland, CA 95695	10-02-79
103. Jeanette Cool 30 Langton San Francisco, CA 94103	10-02-79
104. Marie Pnazeł 2208 Knob Creek Rd. Johnson City, TN 37601	10-02-79
105. James V. Keilman Moon Lake Volunteer Fire Dept. P.O. Box 914, Moon Lake Estates New Port Richey, FL 33552	10-02-79

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106. Edward E. Rice 1819 Lagoon View Drive Tiburon, CA 94920	10-02-79
107. William Dreskin 245 Los Angeles Boulevard San Anselmo, CA 94960	10-02-79
108. Sherman Stein 811 Oak Avenue Davis, CA 95616	10-02-79
109. John M. Bailey, Jr. 625 Huntleigh Drive Lafayette, CA 94549	10-02-79
110. John F. Doherty 4438-1/2 Leeland Houston, TX 77023	10-02-79
111. Ethelyn Brooks 727 Saddlerock Wenatchee, WA 98801	10-02-79
112. Mike Barrett Concerned Texan and Opponent to Nuclear Power and Waste 1505 West Third Avenue Corsicano, TX 75110	10-02-79
113. William N. Hedeman, Jr., Director Office of Environmental Review U.S. Environmental Protection Agency Washington, DC 20460	09-27-79
114. Myra A. Cypser 5743 Leverett Court Alexandria, VA 22311	10-02-79
115. Albert Bates, Project Director PLENTY Shutdown Project 156 Drakes Lane Summertown, TN 38483	10-04-79
116. Ellery W. Newton American Small Farm Institute Mercer Island, WA 98040	10-04-79
117. Naomi M. Proett 1200 Norfolk Avenue W. Sacramento, CA 95691	10-04-79
118. Mrs. Robert Lawton 1403 Randolph Circle Tallahassee, FL 32312	10-04-79

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119. Frank Jantlick 1306 Driftwood Drive New Port Richey, FL 33552	10-04-79
120. Jim McGarr 303 E. College Street Clinton, MS 39056	10-04-79
121. Zelia Jensen, R.N. (Duplicate of 88) Route 1, Box 86A Grandview, TN 37337	10-04-79
122. William J. McClusky (Duplicate of 86) 4317 Rhodes Memphis, TN 38111	10-04-79
123. Michael Griffin 2885 Naples Drive Winter Park, FL 32789	10-04-79
124. Edwin Wiggin Atomic Industrial Forum, Inc. 7101 Wisconsin Avenue Washington, DC 20014	10-04-79
125. Lynne Bowman 4030 Montwood Drive Dallas, TX 75229	10-04-79
126. Robert Leverant P.O. Box 9444 Berkeley, CA 94709	10-04-79
127. Louise Wilson 235 Churchill Avenue Palo Alto, CA 94301	10-04-79
128. Dorothy Newmeyer Perkins Biology Department Stanford University Stanford, CA 94305	10-04-79
129. Mrs. W. W. Schaefer, Chairman Radioactive Waste Management Study Committee Lake Michigan Federation c/o 3741 Koehler Drive Sheboygan, WI 53081	10-04-79
130. R. Marriner Orum 2389 Floral Hill Drive Eugene, OR 97403	10-04-79
131. Matt Edwards 3743 Texas Street San Diego, CA 92104	10-05-79

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132. Mrs Victor N. Tolpegin 11 Eugene Street Mill Valley, CA 94941	10-05-79
133. Mrs. C. R. Nicewonger 80 Codornices Road Berkeley, CA 94708	10-05-79
134. W. R. Flores 5210 Wilson Drive Riverside, CA 92509	10-05-79
135. Winifred A. Many 1063 Euclid Avenue Berkeley, CA 94708	10-05-79
136. Emory L. Ellis P.O. Box 195 Kernville, CA 93238	10-05-79
137. (No name given)	10-05-79
138. Lois and Harold Mills 7660 Morella Avenue North Hollywood, CA 91605	10-05-79
139. Rep. W. J. Tauzin Louisiana House of Representatives Committee on Natural Resources Box 44012, Capitol Station Baton Rouge, LA 70804	10-05-79
140. Cindy and Arthur Scott 8 Athlone Way Menlo Park, CA 94025	10-05-79
141. Julia Corliss P.O. Box 247 Fairfax, CA 94930	10-05-79
142. Phyllis Luckman 668 Fairmount Avenue Oakland, CA 94611	10-05-79
143. H. F. Longway 905 Camelia Avenue Roseville, CA 95678	10-05-79
144. Walbridge J. Powell 4314 Island Crest Way Mercer Island, WA 98040	10-05-79
145. David R. Wallace 4289 A Piedmont Avenue Oakland, CA 94611	10-05-79

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146. Ariadne Rogers Wright 2206 Spring Creek Drive Austin, TX 78704	10-05-79
147. Robert E. Evans 468 W. Melrose, Apt. 351 Chicago, IL 60657	10-05-79
148. Mrs. Barbara Yendle 939 S. Royall Palestine, TX 75801	10-05-79
149. William Klitz 21321/2 Derby Berkeley, CA 94705	10-05-79
150. Donald J. Purinton 1312 Apache Drive Richardson, TX 75080	10-05-79
151. Karl Knute Kresie 555C Stanford Avenue Palo Alto, CA 94306	10-05-79
152. D. G. Mason, Programs Director Fuel and Waste Management Rockwell International Atomics International Division Energy Systems Group 8900 DeSoto Avenue Canoga Park, CA 91304	10-05-79
153. Dabney K. Johnson Rt. 3, Wheeler Rd. Apt. 22 Louisville, TN 37777	10-05-79
154. Lowenstein, Newman, Reis, Axelrad and Toll 1025 Connecticut Avenue, N.W. Washington, DC 20036	10-09-79
155. Mrs. Wendell W. Ray 8755 Junipero Avenue Atascadero, CA 93422	10-09-79
156. Boyce R. Wells Environmental Review Coordinator Commonwealth of Kentucky Department for Natural Resources and Environmental Protection Office of the Secretary Office of Policy and Program Analysis Capital Plaza Tower Frankfort, KY 40601	10-09-79

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157. Frances Pavlovic 111 Datonia Bellaire, TX 77401	10-09-79
158. Norman Gelfand 5640 S. Ellis Avenue Chicago, IL 60637	10-09-79
159. Mr. Wendell W. Ray 8755 Junipero Avenue Atascadero, CA 93422	10-09-79
160. Larry Gaffin First Congregational Church S. Palouse and E. Alder Street Walla Walla, WA 99362	10-09-79
161. John C. Lanigan, Jr. Department of Geology Kent State University Kent, OH 44242	10-09-79
162. Mary A. Stewart 6212 S. El Camino Drive Tempe, AZ 85283	10-10-79
163. Bennie Wooten 27087 Twenty Mule Team Road Boron, CA 93516	10-10-79
164. Murray Solomon SOLCO 11 Vineyard Road Newton, MA 02159	10-10-79
165. Florence Klinger 072 Green Street Martinez, CA 94553	10-10-79
166. Martin E. Fate, Jr. Executive Vice President Public Service Company of Oklahoma P.O. Box 201 Tulsa, OK 74102	10-10-79
167. Jan Clucas, Chairman Santa Lucia Chapter Sierra Club 1727 Corralitos Avenue San Luis Obispo, CA 93401	10-10-79
168. John E. Arthur, Chief Engineer Rochester Gas and Electric Corporation 89 East Avenue Rochester, NY 14649	10-10-79
169. Charity Cone 3000 Pecho Valley Road Los Aros, CA 93402	10-10-79

<u>COMMENTER</u>	<u>DATE RECEIVED</u>
170. Mrs. Edith W. Liu 32 Parkside Drive Davis, CA 95616	10-10-79
171. Robert D. Postar 359 Half Moon Lane #314 Daly City, CA 94015	10-10-79
172. C. R. Nicewonger (Additional comments to 133) 80 Codornices Road Berkeley, CA 94708	10-10-79
173. Mrs. Edith M. Hoyt 3316 Villanova Dallas, TX 75225	10-10-79
174. John Castleberry 2400 Lawndale Plano, TX 75023	10-10-79
175. C. E. Glover Rt. 2 Box 319-A Palestine, TX 75801	10-10-79
176. Kathleen Jensen 122 Cypress Street Alameda, CA 94501	10-10-79
177. Clark L. Moore Resource Consultant 1180 Ash Street Arroyo Grande, CA 93420	10-10-79
178. John B. Gordon, PE 1007 Green Meadow Round Rock, TX 78664	10-10-79
179. David Woolson Rt. 1 Box 26 Medical Lake, WA 99022	10-10-79
180. Jack C. Wells Central and Southwest Services, Inc. 2700 One Main Place Dallas, TX 75250	10-11-79
181. D. S. Webster, Deputy Director Argonne National Laboratory 9700 South Cass Avenue Argonne, IL 60439	10-11-79
182. W. N. Thomas, Vice President Fuel Resources Virginia Electric and Power Company Richmond, VA 23261	10-11-79

<u>COMMENTER</u>	<u>DATE RECEIVED</u>
183. R. A. Langley, Jr. 80 Lansdale Avenue San Francisco, CA 94127	10-11-79
184. Saralee W. Terry, Resource Analyst (Duplicate of 39) Tennessee State Planning Office 660 Capitol Hill Building 301 Seventh Avenue, North Nashville, TN 37219	10-11-79
185. Shelley J. Hegge P.O. Box 10307 Anchorage, AK 99511	10-11-79
186. Marion Stuart 2428 Jefferson Street Berkeley, CA 94703	10-16-79
187. Murray Soloman (Duplicate of 164) SOLCO 11 Vineyard Road Newton, MA 02159	10-12-79
188. Boyd Weeks 27000 N. HWY. 1 Fort Bragg, CA 95437	10-16-79
189. Roy G. Post, Professor The University of Arizona College of Engineering Department of Nuclear Engineering Tucson, AZ 85721	10-16-79
190. Donald J. Stierman Assistant Professor of Geophysics University of California Riverside, California 92521	10-16-79
191. W. W. Schaefer, M.D. Chairman, Committee on Environmental Health The State Medical Society of Wisconsin 330 East Lakeside Street P.O. Box 1109 Madison, WI 53701	10-16-79
192. No name	10-16-79
193. Nancy N. Kroe 3711 West Tilden Place Seattle, WA 98199	10-16-79
194. Linda Hoyt 172 A. Downey Street San Francisco, CA 94117	10-16-79

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195. Dr. and Mrs. Jack M. Troy The Whiting Clinic, Inc. 2075 Indianapolis Blvd. Whiting, IN 46394	10-16-79
196. M. Peter Lanahan, Jr. First Deputy Commissioner State of New York Department of Environmental Conservation Albany, NY 12233	10-16-79
197. Robert A. Phillips 263 Via Cima Ct. Danville, CA 94526	10-16-79
198. M. T. Johnson, General Manager Advanced Energy Systems Division Westinghouse Electric Corporation Box 10864 Pittsburgh, PA 15236	10-18-79
199. S. A. Gray, Jr. 41 E. Liberty Street Waynesboro, GA 31564	10-18-79
200. Phillip R. Compton NASA	10-18-79
201. Frank L. Parker Vanderbilt University Nashville, TN 37235	10-18-79
202. Charles Custard, Director Office of Environmental Affairs Department of Health, Education and Welfare Office of the Secretary Washington, DC 20201	10-18-79
203. James Sharp 307 Granville Rd. Chapel Hill, NC 27514	10-19-79
204. Charleston Chapter of the Palmetto Alliance P.O. Box 582 Charleston, SC 29402	10-19-79
205. Mrs. Helen M. Serenka 319 Los Pinos Way San Jose, CA 75248	10-21-79
206. Linda Oshman 7614 Lairds Lane Dallas, TX 75248	10-21-79
207. Lawrence E. Davis 735 Monssex Parkway Dallas, TX 75224	10-21-79

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208. John B. Martin, Director Division of Waste Management U.S. Nuclear Regulatory Commission Washington, DC 20555	10-29-79
209. Watson Totus, Chairman Yakima Tribal Council Confederated Tribes and Bands Yakima Indian Nation P.O. Box 151 Toppenish, WA 98948	10-29-79
210. Sara Morabito 909 University AV 10 Los Gatos, CA 95030	10-29-79
211. Bryan L. Baker Education and Research Committee Mockingbird Alliance 900 Lovett Blvd.--Suite 207 Houston, TX 77006	10-29-79
212. James R. Schofield 2440A Fulton San Francisco, CA 94118	10-29-79
213. Thomas G. Griffith Griffith & Brister Suite 6F, Lubbock National Building Lubbock, TX 79401	11-01-79
214. Michael Gersick, Deputy Director Department of Conservation Division of Mines and Geology Division of Oil and Gas Sacramento, CA 93814	11-08-79
215. Paul F. Levy, Director Arkansas Department of Energy 3000 Kavanaugh Little Rock, AR 72205	11-16-79
216. Merry Jo Hill Life of the Land 404 Piikoi Street Honolulu, HI 96814	11-19-79
217. Emilio E. Varanini, III Energy Resources Conservation and Development Commission 1111 Howe Avenue Sacramento, CA 95825	11-19-79
218. James Rashlin U.S. Department of the Interior Office of the Secretary Washington, DC 20240	11-20-79
219. National Research Council Commission on Natural Resources 2101 Constitution Avenue Washington, DC 20418	05-18-80

<u>COMMENTS</u>	<u>DATE RECEIVED</u>	<u>LETTER NUMBER</u>
John E. Arthur, Chief Engineer Rochester Gas and Electric Corporation 89 East Avenue Rochester, NY 14649	10-10-79	168
John M. Bailey, Jr. 625 Huntleigh Drive Lafayette, CA 94549	10-02-79	109
Bryan L. Baker Education and Research Committee Mockingbird Alliance 900 Lovett Blvd.--Suite 207 Houston, TX 77006	10-29-79	211
Mike Barrett Concerned Texan and Opponent to Nuclear Power and Waste 1505 West Third Avenue Corsicano, TX 75110	10-02-79	112
Albert Bates, Project Director PLENTY Shutdown Project 156 Drakes Lane Summertown, TN 38483	10-04-79	115
Mr. R. C. Baxter Allied-General Nuclear Services P.O. Box 847 Barnwell, SC 29812	05-31-79	1
Linda Baziuk Route 3, Box 438 Yadkinville, NC 27055	09-20-79	74
Mr. W. P. Bebbington 905 Whitney Drive Aiken, SC 29801	07-03-79	6
Leo Benson, III 1274 Sledge Avenue Memphis, TN 38104	09-21-79	84
Lynne Bowman 4030 Montwood Drive Dallas, TX 75229	10-04-79	125
Ms. Judith Y. Brachman Administering Officer State Clearinghouse 30 East Broad Street 39th Floor Columbus, OH 43215	07-17-79	35
Mr. Ted Breitmayer 148 Draeger Drive Moraga, CA 94556	07-10-79	11

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Elizabeth Briggs 10 Bliss Avenue Woodland, CA 95695	10-02-79	102
Bob Brister 4308 Rhodes Avenue Memphis, TN 38111	09-20-79	78
David K. Brooks P.O. Box 16929 Jackson, MS 39206	09-20-79	71
Ethelyn Brooks 727 Saddlerock Wenatchee, WA 98801	10-02-79	111
Yvonne Brunton 120 Leatnice Lane Vicksburg, MS 39170	09-26-79	101
J. R. Buford 237 Leland Way Hanford, CA 93230	09-12-79	59
Beth E. Buzzard 2016 E. State Avenue Olympia, WA 98506	09-26-79	93
Louis B. Carrick 365 Beechwood Drive Athens, GA 30606	09-18-79	63
John Castleberry 2400 Lawndale Plano, TX 75023	10-10-79	174
Jan Clucas, Chairman Santa Lucia Chapter Sierra Club 1727 Corralitos Avenue San Luis Obispo, CA 93401	10-10-79	167
Phillip R. Compton NASA	10-18-79	200
Charity Cone 3000 Pecho Valley Road Los Aros, CA 93402	10-10-79	169
Jeanette Cool 30 Langton San Francisco, CA 94103	10-02-79	103
Julia Corliss P.O. Box 247 Fairfax, CA 94930	10-05-79	141

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Mr. Ben Cornwell 1795 Magnolia Tree Germantown, TN 38138	09-20-79	72
Mr. T. J. Crossley 16 Loma Linda Lane Lakeland, FL 33803	07-11-79	20
Myra A. Cypser 5743 Leverett Court Alexandria, VA 22311	10-02-79	114
Lawrence E. Davis 735 Monssex Parkway Dallas, TX 75224	10-21-79	207
Robert F. Davis 2706 N. Chamberlain Chattanooga, TN 37406	09-19-79	66
Mr. Gerald R. Day, Executive Director Illinois Commission on Atomic Energy Lincoln Tower Plaza 524 South Second Street--Room 415 Springfield, IL 62706	08-15-79	50
David DeBus, PhD. 425 Bonair Street La Jolla, CA 92037	09-20-79	73
John Dilday 200 Hillsboro St. Cary, NC 27511	09-07-79	56
John F. Doherty 4438-1/2 Leeland Houston, TX 77023	10-02-79	110
Mr. William P. Dornsife, P.E. Bureau of Radiation Protection Department of Environmental Resources P.O. Box 2063 Harrisburg, PA 17120	07-03-79	7
William Dreskin 245 Los Angeles Boulevard San Anselmo, CA 94960	10-02-79	107
Agnes S. Easterly 1100 University Street Seattle, WA 98101	09-19-79	65
Matt Edwards 3743 Texas Street San Diego, CA 92104	10-05-79	131
Emory L. Ellis P.O. Box 195 Kernville, CA 93238	10-05-79	136

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Robert E. Evans 468 W. Melrose, Apt. 351 Chicago, IL 60657	10-05-79	147
Martin E. Fate, Jr. Executive Vice President Public Service Company of Oklahoma P.O. Box 201 Tulsa, OK 74102	10-10-79	166
Chris Findlay 235 Stone Avenue Lexington, KY 40508	09-25-79	89
W. R. Flores 5210 Wilson Drive Riverside, CA 92509	10-05-79	134
Mr. Edward Gabriel Council of Energy Resource Tribes One Thousand Connecticut Avenue, N.W. Suite 610 Washington, DC 20036	07-11-79	18
Harry A. Galliher 250 West Artesia Pomona, CA 91768	09-24-79	85
Mr. W. S. Geiger 741 Woodhill Drive Lakeland, FL 33808	07-10-79	13
Norman Gelfand 5640 S. Ellis Avenue Chicago, IL 60637	10-09-79	158
Michael Gersick, Deputy Director Department of Conservation Division of Mines and Geology Division of Oil and Gas Sacramento, CA 93814	11-08-79	214
C. E. Glover Rt. 2 Box 319-A Palestine, TX 75801	10-10-79	175
John B. Gordon, PE 1007 Green Meadow Round Rock, TX 78664	10-10-79	178
Larry Gaffin First Congregational Church S. Palouse and E. Alder Street Walla Walla, WA 99362	10-09-79	160
S. A. Gray, Jr. 41 E. Liberty Street Waynesboro, GA 31564	10-18-79	199

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Paul Grier 10460 S.W. 111 Street Miami, FL 33176	09-21-79	80
Michael Griffin 2885 Naples Drive Winter Park, FL 32789	10-04-79	123
Thomas G. Griffith Griffith & Brister Suite 6F, Lubbock National Building Lubbock, TX 79401	11-01-79	213
Carl Groff 200 Bolinas RD #81 Fairfax, CA 94930	09-12-79	61
Shelley J. Hegge P.O. Box 10307 Anchorage, AK 99511	10-11-79	185
Edward Hennelly Savannah River Laboratory Aiken, SC 29801	08-30-79	54
Merry Jo Hill Life of the Land 404 Piikoi Street Honolulu, HI 96814	11-19-79	216
Mrs. Irma Ragan Holland 2210 Creston Road Raleigh, NC 27608	09-26-79	99
Mrs. Edith M. Hoyt 3316 Villanova Dallas, TX 75225	10-10-79	173
Linda Hoyt 172 A. Downey Street San Francisco, CA 94117	10-16-79	194
Mr. Daniel Hunt National Science Foundation Washington, DC 20550	07-12-79	27
Mr. J. B. Jackson, Jr. Administrator Commonwealth of Virginia 903 Ninth Street Office Building Richmond, VA 23219	07-19-79	37
Frank Jantlick 1306 Driftwood Drive New Port Richey, FL 33552	10-04-79	119
Kathleen Jensen 122 Cypress Street Alameda, CA 94501	10-10-79	176

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Zelia Jensen, R.N. Route 1, Box 86A Grandview, TN 37337	09-25-79 10-04-79	88, 121
Dabney K. Johnson Rt. 3, Wheeler Rd. Apt. 22 Louisville, TN 37777	10-05-79	153
M. T. Johnson, General Manager Advanced Energy Systems Division Westinghouse Electric Corporation Box 10864 Pittsburgh, PA 15236	10-18-79	198
Edward L. Jones Route 4, Berry Road Taylors, SC 29687	09-20-79	77
Daniel Kaminsky Savannah River Galleries 309 Bull Street Savannah, GA 31401	09-24-79	87
James V. Keilman Moon Lake Volunteer Fire Dept. P.O. Box 914, Moon Lake Estates New Port Richey, FL 33552	10-02-79	105
Mr. Paul Kiepe 2141 1st Avenue S. Payette, ID 83661	07-11-79	15
Marjorie King Carolina Nutrition Corner, Inc. 2514-A. E. North Street, Ext. Burns Corner (E. North St. & Pelham Rd) Greenville, SC 29607	09-21-79	83
J. H. Kittel Argonne National Laboratory 9700 South Cass Avenue Argonne, IL 60439	08-14-79	49
Florence Klinger 072 Green Street Martinez, CA 94553	10-10-79	165
William Klitz 21321/2 Derby Berkeley, CA 94705	10-05-79	149
Karl Knute Kresie 555C Stanford Avenue Palo Alto, CA 94306	10-05-79	151
Nancy N. Kroe 3711 West Tilden Place Seattle, WA 98199	10-16-79	193

<u>COMMENTS</u>	<u>DATE RECEIVED</u>	<u>LETTER NUMBER</u>
Mr. Robert K. Kunita 4234 Lake Ridge Drive Raleigh, NC 27604	07-11-79	21
R. A. Langley, Jr. 80 Lansdale Avenue San Francisco, CA 94127	10-11-79	183
John C. Lanigan, Jr. Department of Geology Kent State University Kent, OH 44242	10-09-79	161
L. E. Lantaff, I.I.C U.S. Department of Agriculture Food Safety and Quality Service Meat and Poultry Inspection Program 1400 S. Green Street, P.O. Box 744 Henderson, KY 42420	09-20-79	69
Charles W. Larimer, Jr. 1308 60th St. Ensley Birmingham, AL 35228	09-26-79	95
Harold V. Larson (Addresses wrong EIS) Battelle, Pacific Northwest Laboratories P.O. Box 999 Richland, WA 99352	08-14-79	48
Ermont M. Lawrence 5241 S.E. 115th Street Bellevue, FL 32620	09-10-79	57
Mrs. Robert Lawton 1403 Randolph Circle Tallahassee, FL 32312	10-04-79	118
Fred L. Lemon Highway 90 and Church Street Ocean Springs, MS 39564	09-26-79	92
Robert Leverant P.O. Box 9444 Berkeley, CA 94709	10-04-79	126
Paul F. Levy, Director Arkansas Department of Energy 3000 Kavanaugh Little Rock, AR 72205	11-16-79	215
Mrs. Lawrence Lewis 3206 Columbia Parkway Cincinnati, OH 45206	08-09-79	47
Mrs. Lawrence Lewis (Additional comments to 47) 3206 Columbia Parkway Cincinnati, OH 45206	08-17-79	51

<u>COMMENTER</u>	<u>DATE RECEIVED</u>	<u>LETTER NUMBER</u>
Mr. Marvin I. Lewis 6504 Bradford Terrace Philadelphia, PA 19149	07-13-79	30
Mr. Paul W. Levy Brookhaven National Laboratory Associated Universities, Inc. Upton, NY 11973	07-12-79	25
Mrs. Edith W. Liu 32 Parkside Drive Davis, CA 95616	10-10-79	170
Roland Livernois Livernois Enterprise 1231 Kenyon Road Lake City, GA 30252	09-19-79	67
Mr. William A. Lochstet 119 E. Aaron Dr. State College, PA 16801	07-06-79	8
H. F. Longway 905 Camelia Avenue Roseville, CA 95678	10-05-79	143
Lowenstein, Newman, Reis, Axelrad and Toll 1025 Connecticut Avenue, N.W. Washington, DC 20036	10-09-79	154
Phyllis Luckman 668 Fairmount Avenue Oakland, CA 94611	10-05-79	142
Brandt Mannchen 4055 South Braeswood #303 Houston, TX 77025	09-12-79	62
D. G. Mason, Programs Director Fuel and Waste Management Rockwell International Atomics International Division Energy Systems Group 8900 DeSoto Avenue Canoga Park, CA 91304	10-05-79	152
Winifred A. Many 1063 Euclid Avenue Berkeley, CA 94708	10-05-79	135
Mr. J. A. McBride Box 192 Merrifield, VA 22116	07-23-79	38
William J. McClusky 4317 Rhodes Memphis, TN 38111	09-24-79 10-04-79	86, 122

<u>COMMENTER</u>	<u>DATE RECEIVED</u>	<u>LETTER NUMBER</u>
Jim McGarr 303 E. College Street Clinton, MS 39056	10-04-79	120
Ms. Edith M. McKee 416 Maple Street Winnetka, IL 60093	07-10-79	12
Mr. W. J. Mecham Argonne National Laboratory 9700 South Cass Avenue Argonne, IL 60439	06-27-79	5
Lois and Harold Mills 7660 Morella Avenue North Hollywood, CA 91605	10-05-79	138
Lorane H. Minis 4142 Amsterdam Circle Savannah, GA 31405	09-25-79	90
Mr. Robert H. Moen 6602 Tam O'Shanter Drive San Jose, CA 95120	07-12-79	26
Clark L. Moore Resource Consultant 1180 Ash Street Arroyo Grande, CA 93420	10-10-79	177
Mr. J. P. Moore 6107 Main Avenue Tampa, FL 33611	09-21-79	81
Sara Morabito 909 University AV 10 Los Gatos, CA 95030	10-29-79	210
James Mulloy Department of Water and Power 111 North Hope Street Los Angeles, CA 90012	08-08-79	45
National Research Council Commission on Natural Resources 2101 Constitution Avenue Washington, DC 20418	05-18-80	219
Mrs. Marge Nejedly 109 W. Kennedy Drive Holiday, FL 33590	09-26-79	91
Dorothy Newmeyer Perkins Biology Department Stanford University Stanford, CA 94305	10-04-79	128

<u>COMMENTER</u>	<u>DATE RECEIVED</u>	<u>LETTER NUMBER</u>
Ellery W. Newton American Small Farm Institute Mercer Island, WA 98040	10-04-79	116
Mrs. C. R. Nicewonger 80 Codornices Road Berkeley, CA 94708	10-05-79	133
C. R. Nicewonger (Additional comments to 133) 80 Codornices Road Berkeley, CA 94708	10-10-79	172
Ms. Emily Neary State of Vermont Office of the Governor State A-95 Clearinghouse Montpelier, VT 05602	07-11-79	19
Paulette Oakes 1118 Ingra Anchorage, AK 99501	09-12-79	60
Mr. Donald Orth Savannah River Laboratory P.O. Box A Aiken, SC 29801	07-10-79	10
R. Marriner Orum 2389 Floral Hill Drive Eugene, OR 97403	10-04-79	130
Linda Oshman 7614 Lairds Lane Dallas, TX 75248	10-21-79	206
Charleston Chapter of the Palmetto Alliance P.O. Box 582 Charleston, SC 29402	10-19-79	204
Frank L. Parker Vanderbilt University Nashville, TN 37235	10-18-79	201
Charles L. Pater Belden Corporation 2000 South Cass Avenue Geneva, IL 60134	08-21-79	53
Frances Pavlovic 111 Datonia Bellaire, TX 77401	10-09-79	157
Robert A. Phillips 263 Via Cima Ct. Danville, CA 94526	10-16-79	197

<u>COMMENTS</u>	<u>DATE RECEIVED</u>	<u>LETTER NUMBER</u>
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Mr. Wendell W. Ray 8755 Junipero Avenue Atascadero, CA 93422	10-09-79	159

<u>COMMENTS</u>	<u>DATE RECEIVED</u>	<u>LETTER NUMBER</u>
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James R. Schofield 2440A Fulton San Francisco, CA 94118	10-29-79	212
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Cindy and Arthur Scott 8 Athlone Way Menlo Park, CA 94025	10-05-79	140
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James Sharp 307 Granville Rd. Chapel Hill, NC 27514	10-19-79	203
David Shields 419 Oakdale Rd. N.E. Atlanta, GA 30307	09-19-79	75
Murray Solomon SOLCO 11 Vineyard Road Newton, MA 02159	10-10-79 10-12-79	164, 187

<u>COMMENTS</u>	<u>DATE RECEIVED</u>	<u>LETTER NUMBER</u>
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Sherman Stein 811 Oak Avenue Davis, CA 95616	10-02-79	108
Mary A. Stewart 6212 S. El Camino Drive Tempe, AZ 85283	10-10-79	162
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Marion Stuart 2428 Jefferson Street Berkeley, CA 94703	10-16-79	186
W. N. Thomas, Vice President Fuel Resources Virginia Electric and Power Company Richmond, VA 23261	10-11-79	182
Mrs Victor N. Tolpegin 11 Eugene Street Mill Valley, CA 94941	10-05-79	132
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<u>COMMENTS</u>	<u>DATE RECEIVED</u>	<u>LETTER NUMBER</u>
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Mr. Derek Wallentinsen Albuquerque Group Sierra Club 3131 Quincy NE Albuquerque, NM 87110	07-13-79	28
Walter E. Wallis Wallis Engineering 1954--R Old Middlefield Way Mountain View, CA 94043	09-18-79	64
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Boyd Weeks 27000 N. HWY. 1 Fort Bragg, CA 95437	10-16-79	188
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Louise Wilson 235 Churchill Avenue Palo Alto, CA 94301	10-04-79	127
David Woolson Rt. 1 Box 26 Medical Lake, WA 99022	10-10-79	179
Bennie Wooten 27087 Twenty Mule Team Road Boron, CA 93516	10-10-79	163
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<u>COMMENTER</u>	<u>DATE RECEIVED</u>	<u>LETTER NUMBER</u>
Mr. Joel Yellin Harvard University John F. Kennedy School of Government 79 Boylston Street Cambridge, MA 02138	07-25-79	40
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Kenneth Yonovitz 4206 Washington Road West Palm Beach, FL 33405	09-21-79	82
Lyle L. Zahn, Jr., Manager Spent Fuel Services Operation General Electric Company 175 Curtner Avenue San Jose, CA 95125	09-10-79	58
District of Columbia:		
Mr. Sidney R. Galler United States Department of Commerce The Assistant Secretary for Science and Technology Washington, DC 20230	07-11-79	23
James Rashlin U.S. Department of the Interior Office of the Secretary Washington, DC 20240	11-20-79	218
John B. Martin, Director Division of Waste Management U.S. Nuclear Regulatory Commission Washington, DC 20555	10-29-79	208
Charles Custard, Director Office of Environmental Affairs Department of Health, Education and Welfare Office of the Secretary Washington, DC 20201	10-18-79	202
William N. Hedeman, Jr., Director Office of Environmental Review U.S. Environmental Protection Agency Washington, DC 20460	09-27-79	113
Florida:		
J. S. Sherman Department of Environmental Regulation Twin Towers Office Building 2600 Blair Stone Road Tallahassee, FL 32301	07-31-79	42

<u>COMMENTER</u>	<u>DATE RECEIVED</u>	<u>LETTER NUMBER</u>
Ron Fahs, Director (Duplicate of 42) Intergovernmental Coordination State of Florida Office of the Governor Tallahassee, FL 32301	09-19-79	68
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George Busbee Office of the Governor Atlanta, GA 30334	09-26-79	97
Mr. Arne O. Kauranen Regional Engineer Federal Energy Regulatory Commission Regional Office 730 Peachtree Street, N.E. Atlanta, GA 30308	06-27-79	4
Idaho:		
Ms. Pam Demo-Rybus Division of Budget, Policy Planning and Coordination Executive Office of the Governor State Clearinghouse State House Boise, ID 83720	07-10-79	14
Kansas:		
Mr. Paul DeGaeta Department of Administration Division of State Planning and Research Fourth Floor--Mills Building 109 W. 9th Topeka, KS 66612	07-16-79	31
Kentucky:		
Boyce R. Wells Environmental Review Coordinator Commonwealth of Kentucky Department for Natural Resources and Environmental Protection Office of the Secretary Office of Policy and Program Analysis Capital Plaza Tower Frankfort, KY 40601	10-09-79	156
Louisiana:		
Rep. W. J. Tauzin Louisiana House of Representatives Committee on Natural Resources Box 44012, Capitol Station Baton Rouge, LA 70804	10-05-79	139

<u>COMMENTER</u>	<u>DATE RECEIVED</u>	<u>LETTER NUMBER</u>
Mr. B. Jim Porter Office of Conservation P.O. Box 14690 Baton Rouge, LA 70808	07-16-79	33
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Nebraska:		
Mr. Terry Kubicek Natural Resources Coordinator State Office of Planning & Programming Box 94601, State Capital, Room 1321 Lincoln, NB 68509	06-27-79	3
Nevada:		
Mr. Mike Nolan Governor's Office of Planning Coordination Capitol Complex Carson City, NV 89710	07-12-79	24
New Jersey:		
Mr. Lawrence Schmidt, Chief Office of Environmental Review Department of Environmental Protection P.O. Box 1390 Trenton, NJ 08625	07-16-79	32
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M. Peter Lanahan, Jr. First Deputy Commissioner State of New York Department of Environmental Conservation Albany, NY 12233	10-16-79	196

<u>COMMENTER</u>	<u>DATE RECEIVED</u>	<u>LETTER NUMBER</u>
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Herbert L. Harper Executive Director and State Historic Preservation Officer Tennessee Department of Conservation Tennessee Historical Commission 4721 Trousdale Drive Nashville, TN 37220	09-20-79	70
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Mr. Thomas M. Webb, Director Natural Resources Planning Tennessee State Planning Office 660 Capitol Hill Building 301 Seventh Avenue, North Nashville, TN 37219	07-23-79	39
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Mr. Donald E. Harley (Addresses wrong EIS) Office of the Governor Executive Office Building 411 West 13th Street Austin, TX 78701	07-11-79	22
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Lee Sherman Dreyfus Governor, State of Wisconsin State Capitol Madison, WI 53702	08-06-79	43
W. W. Schaefer, M.D. Chairman, Committee on Environmental Health The State Medical Society of Wisconsin 330 East Lakeside Street P.O. Box 1109 Madison, WI 53701	10-16-79	191

<u>COMMENTS</u>	<u>DATE RECEIVED</u>	<u>LETTER NUMBER</u>
Wyoming: Dick Hartman State Planning Coordinator Wyoming Executive Department Cheyenne, WY 82001	07-27-79	41

APPENDIX B

INDEX FOR COMMENT LETTERS

APPENDIX B

INDEX FOR COMMENT LETTERS

Appendix B identifies the topic area(s) discussed in a particular comment letter, as well as the location of the comment in this volume. This appendix is structured such that a list of the letter numbers is presented along with the topic areas (under Policy and Technical categories) and page numbers(s) on which the letter has been cited and for which responses have been provided.

Letter	Topic Area	Pages
1	Costs	195
2	Waste Management Program	6, 10
	Siting Issues	16
	Foreign Wastes	19
	Attribution of Costs/Risks	22
	Organization and Presentation	28
	Waste Management Operations	139, 178
	Geologic Considerations	206
3	Waste Management Program	7, 10
	Licensing and the Decision-Making Process	11
	Attribution of Costs/Risks	22
4-FERC	Siting Issues	15
6	Siting Issues	16
	Alternative Disposal Issues	23
	Risk Perspectives	136, 137
	Waste Management Operations	143, 144, 147, 152, 184
	Fuel Cycles	189
	Geologic Considerations	207, 212
	Multibarriers for Disposal	281
	Alternative Disposal Concepts	394
7	Waste Management Program	8
	General Comments	24
	Organization and Presentation	28
	Scope	36
	Consequence Analysis	108
	Socioeconomic/Sociopolitical Issues	290
8	Attribution of Costs/Risks	22
	Radiological Issues	45
	Dose Calculations	114, 117, 119
	Risk Perspectives	132
9	Waste Management Program	8
	Attribution of Costs/Risks	22
	Alternative Disposal Issues	23
	Multibarriers for Disposal	278
	Alternative Disposal Concepts	320, 321
10	Radiological Issues	47
	Geologic Considerations	267
11	Alternative Disposal Issues	23
	Geologic Considerations	206
	Comparative Assessment	301, 303
	Alternative Disposal Concepts	385
12	Waste Management Program	10
	Geologic Considerations	269
	Reference Environments	296
13	Waste Management Program	10
	Organization and Presentation	28
	Dose Calculations	126
	Waste Management Operations	143
	Costs	197
	Geologic Considerations	206

Letter	Topic Area	Pages
14	Scope Costs	36 193
15	Siting Issues General Comments Socioeconomic/Sociopolitical Issues	16 24 292
17	Waste Management Program Licensing and the Decision-Making Process Long Term Storage Radiological Issues Consequence Analysis Geologic Considerations Multibarriers for Disposal Socioeconomic/Sociopolitical Issues	10 11 21 48 108 224 277 283
18	Socioeconomic/Sociopolitical Issues	291
19	Licensing and the Decision-Making Process	12
20	General Comments Socioeconomic/Sociopolitical Issues	24 293
21	Waste Management Program	6, 7, 10
22	Waste Management Program Licensing and the Decision-Making Process Foreign Wastes Fuel Cycle Issues Growth Scenarios	6 12 19 20 204
23-DOC	Alternate Disposal Issues Alternative Disposal Concepts	23, 347, 349, 352, 356, 361, 363, 364, 371
25	Geologic Considerations	268, 269
26	Scope	34
27	Waste Management Program General Comments Waste Management Operations	7, 10 24 186
28	Waste Management Program Licensing and the Decision-Making Process Siting Issues Alternative Disposal Issues Consequence Analysis Dose Calculations Waste Management Operations Comparative Assessment	6, 8, 10 11, 12 15, 16 23 111 114 152 311
29	Waste Management Program General Comments	9 24
30	Waste Management Program Alternative Disposal Issues Radiological Issues Consequence Analysis Dose Calculations Risk Perspectives Waste Management Operations Growth Scenarios	6 23 40, 52 67, 73, 74, 91 116 131 158 204

Letter	Topic Area	Pages
30 (Contd)	Geologic Considerations Resource Requirements	224, 226 295
31	Siting Issues	15
32	Waste Management Program Risk Perspectives Fuel Cycles	9, 10 131 188
33	Waste Management Program	10
34	Waste Management Program Organizaton and Presentation Scope Consequence Analysis Dose Calculations Geologic Consideration Multibarriers for Disposal Resource Requirements Comparative Assessment Alternative Disposal Concepts	10 27, 28 34, 38 62 115 210 272 294 303 339, 377, 384
35	Waste Management Operations Siting Issues Long Term Storage Alternative Disposal Issues General Comments Organization and Presentation Consequence Analysis Waste Management Operations Fuel Cycles Growth Scenarios Geologic Considerations Multibarriers for Disposal Socioeconomic/Sociopolitical Issues Alternative Disposal Concepts	9, 10 16 21 23 24 28 76 147, 152, 172, 176, 178 189 204 257 278 291 326, 336, 399, 400
36	Waste Management Operations Attribution of Costs/Risks Alternative Disposal Issues Multibarriers for Disposal Alternative Disposal Concepts	8 22 23 278 320, 321
37	Licensing and the Decision-Making Process	12
38	Organization and Presentation Scope Consequence Analysis Risk Perspectives Socioeconomic/Sociopolitical Issues Comparative Assessment	28 38 111 128, 133, 137 291 311
39	Siting Issues Fuel Cycle Issues	15 20
40	Radiological Issues Consequence Analysis Risk Perspectives Waste Management Operations Safeguards Growth Scenarios Geologic Considerations	39 67, 72, 75, 89, 91, 92, 108 131 139, 171, 175 199 204, 205 243

<u>Letter</u>	<u>Topic Area</u>	<u>Pages</u>
40 (Contd)	Socioeconomic/Sociopolitical Issues	284, 287, 288
	Alternative Disposal Concepts	320
41	Waste Management Operations	10
	Licensing and the Decision-Making Process	12
	Risk Perspectives	131
	Socioeconomic/Sociopolitical Issues	283
42	Waste Management Program	8
	Attribution of Costs/Risks	22
	General Comments	24
	Waste Management Operations	142
	Growth Scenarios	204, 205
	Geologic Considerations	263
43	Licensing and the Decision-Making Process	12, 13
	Siting Issues	15, 16
	Alternative Disposal Issues	23
	Organization and Presentation	26
	Scope	34, 36
	Radiological Issues	40, 49, 59
	Consequence Analysis	75, 111
	Dose Calculations	119, 122
	Waste Management Operations	169, 178, 180
	Costs	195, 197
	Growth Scenarios	204
	Geologic Considerations	214, 216, 229, 262, 264, 266
	Multibarriers for Disposal	274, 276, 277, 281
	Socioeconomic/Sociopolitical Issues	283, 284, 285, 291
	Resource Requirements	294
	Reference Environment	296
	Alternative Disposal Concepts	316, 318, 319, 323, 334, 378, 381, 382, 383, 387
44	Licensing and the Decision-Making Process	11
45	Waste Management Program	6, 7
	Long Term Storage	21
49	Waste Management Program	10
52	Siting Issues	16
53	Waste Management Program	10
55	Waste Management Program	10
	Long Term Storage	21
	Attribution of Costs/Risks	22
	General Comments	24
	Radiological Issues	46
	Consequence Analysis	109
	Waste Management Operations	139, 140, 149, 155
	Safeguards	202
	Growth Scenarios	204
	Geologic Considerations	206
	Multibarriers for Disposal	279
	Socioeconomic/Sociopolitical Issues	284, 293
56	Long Term Storage	21

Letter	Topic Area	Pages
57	General Comments	24
58	Organizaton and Presentation	28
	Consequence Analysis	67, 76, 79, 111
	Risk Perspectives	137
	Waste Management Operations	149, 170, 178
	Multibarriers for Disposal	273, 281
	Socioeconomic/Sociopolitical Issues	287
	Comparative Assessment	311
	Alternative Disposal Concepts	319, 326, 335, 337, 346, 349, 365, 376, 379, 389, 391, 395, 398, 399, 402, 405
59	General Comments	24
60	General Comments	24
62	Foreign Wastes	19
	Fuel Cycle Issues	20
	Alternative Disposal Issues	23
	Scope	37
	Radiological Issues	40
	Consequence Analysis	64, 65
	Waste Management Operations	139, 146, 147, 171, 172
	Growth Scenarios	204, 205
	Geologic Considerations	252, 264
	Socioeconomic/Sociopolitical Issues	283
	Alternative Disposal Concepts	317
63	Licensing and the Decision-Making Process:	12
	Foreign Wastes	19
	Alternative Disposal Issues	23
	General Comments	24
	Growth Scenarios	204
64	Waste Management Program	10
65	General Comments	24
67	General Comments	24
68	Waste Management Program	8
	Attribution of Costs/Risks	22
	General Comments	24
	Waste Management Operations	142
	Growth Scenarios	204, 205
	Geologic Considerations	263
69	General Comments	24
70	Siting Issues	15
71	Siting Issues	16
72	General Comments	24
	Safeguards	202
73	Geologic Considerations	261
74	Siting Issues	16
	Attribution of Costs/Risks	22
	General Comments	24

<u>Letter</u>	<u>Topic Area</u>	<u>Pages</u>
76	General Comments	24
77	Attribution of Costs/Risks	22
78	Long Term Storage General Comments	21 24
82	General Comments Alternative Disposal Concepts	24 316
83	General Comments	24
84	Alternative Disposal Issues	23
86	Waste Management Program Long Term Storage Alternative Disposal Issues	10 21 23
87	Siting Issues	16
88	Alternative Disposal Issues Alternative Disposal Concepts	23 320, 330, 340, 355, 378, 388, 396, 402
90	General Comments	24
91	General Comments	24
93	Siting Issues General Comments	16 24
94	Siting Issues	16
95	Licensing and the Decision-Making Process	11
96	Waste Management Program Alternative Disposal Issues General Comments Geologic Considerations Reference Environment	7 23 24 206 296
97	Licensing and the Decision-Making Process Siting Issues Fuel Cycle Issues Organization and Presentation Scope Consequence Analysis Risk Perspectives Geologic Considerations Multibarriers for Disposal Alternative Disposal Concepts	12, 13 16 20 26 36 96, 99, 111 137 206, 262, 266, 267 277, 281, 282 316
98	Waste Management Program Licensing and the Decision-Making Process Foreign Wastes General Comments Risk Perspectives Geologic Considerations	8 12 19 24 137 266
99	Siting Issues	16
100	Siting Issues	16

<u>Letter</u>	<u>Topic Area</u>	<u>Pages</u>
102	General Comments	24
103	General Comments	24
104	Radiological Issues	58
105	General Comments	24
106	General Comments	24
108	General Comments Consequence Analysis	24 68
109	General Comments	24
110	General Comments	24
111	Foreign Wastes General Comments	19 24
112	Siting Issues Attribution of Costs/Risks	16 22
113-EPA	Siting Issues Alternative Disposal Issues Organization and Presentation Scope Radiological Issues Consequence Analysis Dose Calculations Risk Perspectives Waste Management Operations Costs Growth Scenarios Geologic Considerations Multibarriers for Disposal Socioeconomic/Sociopolitical Reference Environment Comparative Assessment Alternative Disposal Concepts	15 23 32 38 40, 41, 42, 43, 44, 45, 46, 47, 48, 49, 50, 51, 52, 53, 55, 56, 57, 58 63, 67, 70, 71, 72, 76, 82, 83, 84, 85, 86, 88, 89, 93, 94, 95, 97, 98, 99, 111 113, 117, 118, 119, 121, 123, 125, 126 129, 132, 133, 134, 135, 136 145, 149, 152, 168, 177, 179, 180 192, 194, 196 205 208, 213, 217, 221, 224, 230, 232, 234, 235, 236, 237, 238, 249, 250, 251, 253, 256, 258 271, 273, 275, 277, 279, 280, 281 283, 285, 287, 289 290, 292 297, 298 301 314, 318, 325, 353, 355, 356, 357, 361, 362, 364, 365, 366, 368, 369, 373, 374, 375, 377, 406
114	Consequence Analysis Multibarriers for Disposal Socioeconomic/Sociopolitical Issues	67, 68, 73, 90 281 291, 292

<u>Letter</u>	<u>Topic Area</u>	<u>Pages</u>
115	Siting Issues	16
	Radiological Issues	60
	Geologic Considerations	266
116	Siting Issues	16
	Attribution of Costs/Risks	22
117	General Comments	24
118	General Comments	24
119	Alternative Disposal Issues	23
120	Alternative Disposal Issues	23
121	Alternative Disposal Issues	23
	Alternative Disposal Concepts	320, 330, 340, 355, 378, 388, 396, 402
122	Waste Management Program	10
	Long Term Storage	21
	Alternative Disposal Issues	23
123	Licensing and the Decision-Making Process	12
	Alternative Disposal Issues	23
	General Comments	24
124	Organization and Presentation	28
	Consequence Analysis	111
	Risk Perspectives	128, 132, 137
	Waste Management Operations	170
	Safeguards	199, 201
	Geologic Considerations	217
	Multibarriers for Disposal	274, 281
	Alternative Disposal Concepts	321, 341, 342, 343, 347, 348, 354, 358, 359, 361, 363, 367, 374, 378, 379, 380, 381, 386, 389, 393, 395, 397, 398, 401, 402, 403, 404, 405, 406
125	General Comments	24
126	Scope	36
127	General Comments	24
128	Waste Management Programs	6, 10
	General Comments	24
	Consequence Analysis	64
	Risk Perspectives	131
	Safeguards	199
	Geologic Considerations	208
	Comparative Assessment	303
129	Licensing and the Decision-Making Process	12
	Waste Management Operations	140
	Geologic Considerations	206
	Reference Environment	296
131	General Comments	24

Letter	Topic Area	Pages
133	Waste Management Programs Long Term Storage	10 21
134	General Comments Risk Perspectives	24 137
135	Siting Issues General Comments	16 24
136	Siting Fuel Cycle Issues	16 20
137	General Comments	24
138	General Comments	24
139	Waste Management Programs Licensing and the Decision-Making Process Siting Issues	6 12 15, 16
140	Alternative Disposal Issues General Comments	23 24
141	Alternative Disposal Issues General Comments Geologic Considerations Multibarriers for Disposal	23 24 218 281
142	General Comments Consequence Analysis Waste Management Operations Safeguards Multibarriers for Disposal	24 67, 68 173 199 281
143	General Comments	24
144	Waste Management Program Alternative Disposal Issues Scope Dose Calculations Risk Perspectives Waste Management Operations Geologic Considerations	6 23 35, 37 114 131 162, 173 206
145	General Comments Scope Risk Perspectives	24 36 131
146	General Comments	24
147	Licensing and the Decision-Making Process General Comments Scope Radiological Issues Dose Calculations Risk Perspectives Waste Management Operations Geologic Considerations Comparative Assessment Alternative Disposal Concepts	13 24 35, 37 40, 44 116 128, 131 139, 146, 147 206 303 385
148	Siting Issues	16

<u>Letter</u>	<u>Topic Area</u>	<u>Pages</u>
149	Alternative Disposal Issues	23
	General Comments	24
151	Waste Management Program	7
	Attribution of Costs/Risks	22
152	Waste Management Program	8, 9, 10
153	Waste Management Program	9, 10
	General Comments	24
	Geologic Considerations	262
154	Waste Management Program	6, 10
	Licensing and the Decision-Making Process	13
	Organization and Presentation	28, 29, 31
	Scope	34, 35, 37, 38
	Radiological Issues	43
	Consequence Analysis	67, 68, 76, 83, 85, 91, 92
	Dose Calculations	118, 120, 121
	Risk Perspectives	128, 131, 132, 137
	Waste Management Operations	141, 150, 152, 153, 156, 159, 160, 163, 170, 172, 173, 181, 183
	Fuel Cycles	189
	Costs	193, 194
	Safeguards	201
	Growth Scenarios	204
	Geologic Considerations	207, 209, 211, 212, 214, 215, 216, 218, 222, 227, 228, 229, 233, 234, 235, 236, 240, 241, 245, 246, 247, 248, 249, 250, 256
	Multibarriers for Disposal	274, 275, 277, 281
	Socioeconomic/Sociopolitical Issues	283, 284, 286, 291
	Comparative Assessment	301, 302, 303
	Alternative Disposal Concepts	313, 317, 328, 333, 339, 340, 344, 345, 346, 351, 353, 354, 360, 364, 366, 369, 375, 376, 379, 380, 384, 385, 389, 390, 391, 392, 393, 396, 400, 403, 404, 405
155	Fuel Cycle Issues	20
	Attribution of Costs/Risks	22
	General Comments	24
	Scope	37
156	Licensing and the Decision-Making Process	12
157	Siting Issues	16
	Fuel Cycle Issues	20
	Long Term Storage	21
158	Waste Management Program	10
	Long Term Storage	21
	Alternative Disposal Issues	23
	General Comments	24
159	General Comments	24

Letter	Topic Area	Pages
160	General Comments	24
162	Siting Issues General Comments	16 24
163	Attribution of Costs/Risks	22
164	Waste Management Program Alternative Disposal Issues Geologic Considerations	7, 9, 10 23 268
165	General Comments	24
166	Waste Management Program Radiological Issues Consequence Analysis Dose Calculations Risk Perspectives Geologic Considerations	6, 10 39 90 113 128, 131, 133 206
167	Waste Management Program Foreign Wastes Organization and Presentation Scope Risk Perspectives Growth Scenarios Comparative Assessment Alternative Disposal Concepts	10 19 28, 32 38 136 205 304 315
168	Waste Management Program Consequence Analysis	10 66
169	General Comments	24
170	Risk Perspectives	131
171	General Comments Socioeconomic/Sociopolitical Issues	24 283
172	General Comments	24
173	Siting Issues General Comments	16 24
174	Siting Issues General Comments	16 24
175	Siting Issues	16
177	Waste Management Program Alternative Disposal Issues General Comments	10 23 24
178	Risk Perspectives Geologic Considerations	138 262
179	Siting Issues	16
180	Waste Management Program Siting Issues Alternative Disposal Issues General Comments	6, 10 16 23 24

Letter	Topic Area	Pages
181	Licensing and the Decision-Making Process	13
	Siting Issues	16, 17
	General Comments	24
	Organization and Presentation	26, 27, 28, 32
	Dose Calculations	114
	Waste Management Operations	139, 140, 141, 142, 147, 148, 150, 151, 152, 156, 157, 158, 159, 162, 167, 170, 172
	Fuel Cycles	188
	Safeguards	200
	Geologic Considerations	206, 239
	Multibarriers for Disposal	276
	Socioeconomic/Sociopolitical Issues	283
	Alternative Disposal Concepts	317, 321
182	Waste Management Program	10
	Organization and Presentation	28
183	Waste Management Program	10
184	Siting Issues	15
	Fuel Cycle Issues	20
185	General Comments	24
186	Foreign Wastes	19
	General Comments	24
	Socioeconomic/Sociopolitical Issues	283
187	Waste Management Program	7, 9, 10
	Alternative Disposal Issues	23
	Geologic Considerations	268
190	Geologic Considerations	209, 210
191	Waste Management Program	10
	Licensing and the Decision-Making Process	12
192	Alternative Disposal Issues	23
193	Attribution of Costs/Risks	22
194	Alternative Disposal Issues	23
	General Comments	24
195	Waste Management Program	10
	General Comments	24
	Risk Perspectives	132
196	Waste Management Program	7, 10
	Fuel Cycle Issues	20
	Alternative Disposal Issues	23
	Scope	38
	Consequence Analysis	111
	Dose Calculations	114
	Risk Perspectives	133
197	General Comments	24
	Scope	36
	Risk Perspectives	130
	Geologic Considerations	264

<u>Letter</u>	<u>Topic Area</u>	<u>Pages</u>
198	Waste Management Program	9
	Licensing and the Decision-Making Process	13
	Fuel Cycle Issues	20
	General Comments	24
	Organization and Presentation	28, 32
	Radiological Issues	43, 47
	Consequence Analysis	68, 80, 86
	Risk Perspectives	128, 130
	Fuel Cycles	188
	Geologic Considerations	206
	Multibarriers for Disposal	276
	Socioeconomic/Sociopolitical Issues	291
	Comparative Assessment	301
Alternative Disposal Concepts	394	
199	Waste Management Program	10
	Attribution of Costs/Risks	22
	General Comments	24
200	Alternative Disposal Concepts	406
201	Organization and Presentation	28, 31, 32
	Comparative Assessment	301
202-HEW	Licensing and the Decision-Making Process	11
	Organization and Presentation	31
	Radiological Issues	43
	Consequence Analysis	111
	Dose Calculations	127
203	Waste Management Program	9
204	Licensing and the Decision-Making Process	12
	Siting Issues	16
	General Comments	24
205	Waste Management Program	7, 9, 10
	Attribution of Costs/Risks	22
	General Comments	24
206	Siting Issues	16
	General Comments	24
207	Siting Issues	16
208-NRC	Licensing and the Decision-Making Process	12, 13
	Siting Issues	17
	Organization and Presentation	26, 27, 28, 33
	Scope	34, 35, 38
	Radiological Issues	39, 49, 50, 52, 59,
	Consequence Analysis	61, 62, 64, 65, 66, 69,
		71, 73, 75, 77, 78, 79,
		80, 81, 83, 86, 87, 88,
		93, 100, 101, 102, 103,
		105, 107, 108, 109, 112
Dose Calculations	113, 115, 116, 117, 121,	
	122, 123, 124, 125, 126	
Risk Perspectives	128, 129, 135	

<u>Letter</u>	<u>Topic Area</u>	<u>Pages</u>
208 (Contd)	Waste Management Operations	140, 143, 144, 145, 146, 148, 153, 154, 155, 156, 157, 159, 160, 161, 163, 164, 166, 167, 168, 171, 172, 173, 174, 175, 176, 177, 178, 181, 182, 183, 184, 185, 186, 187
	Costs	190, 193, 197
	Safeguards	200, 202, 203
	Growth Scenarios	204
	Geologic Considerations	206, 208, 212, 213, 214, 219, 220, 221, 222, 223, 224, 225, 226, 227, 228, 229, 230, 232, 234, 236, 237, 238, 239, 240, 241, 243, 244, 246, 247, 249, 250, 252, 253, 254, 255, 259, 260, 261, 263, 264, 265, 270
	Multibarriers for Disposal	271
	Socioeconomic/Sociopolitical Issues	284, 285, 286, 288
	Resource Requirements	294, 295
	Reference Environment	296, 298
	Comparative Assessment	300, 301, 302, 303, 305, 306, 307, 308, 309, 310
	Alternative Disposal Concepts	313, 316, 317, 322, 327, 328, 329, 330, 331, 332, 333, 334, 335, 336, 337, 338, 339, 340, 341, 342, 343, 344, 345, 348, 349, 350, 351, 354, 355, 346, 357, 358, 359, 360, 361, 365, 371, 372, 373, 379, 387, 388
209	Siting Issues	16
	General Comments	24
210	Waste Management Program	10
211	Risk Perspectives	130, 132
	Geologic Considerations	206
212	Alternative Disposal Issues	23
	General Comments	24
213	Alternative Disposal Issues	23
	General Comments	24
	Consequence Analysis	69
	Waste Management Operations	173, 178
	Costs	192
	Geologic Considerations	206, 218
	Socioeconomic/Sociopolitical Issues	283
214	Licensing and the Decision-Making Process	13
	Organization and Presentation	32
	Radiological Issues	43
	Consequence Analysis	90
	Geologic Considerations	226, 229, 231, 239, 242, 251, 255
	Multibarriers for Disposal	281

<u>Letter</u>	<u>Topic Area</u>	<u>Pages</u>
215	Waste Management Program Scope	10 36
216	Alternative Disposal Issues General Comments Scope Costs Socioeconomic/Sociopolitical Issues	23 24 36 198 283
217	Waste Management Program Licensing and the Decision-Making Process Fuel Cycle Issues Organization and Presentation Scope Consequence Analysis Risk Perspectives Costs Comparative Assessment	10 13 20 26, 28, 32 37 61 132, 134, 137 195 302, 304, 310, 311
218-DOI	Waste Management Program Licensing and the Decision-Making Process Siting Issues Foreign Wastes Organization and Presentation Radiological Issues Consequence Analysis Risk Perspectives Waste Management Operations Fuel Cycles Safeguards Geologic Considerations Multibarriers for Disposal Socioeconomic/Sociopolitical Issues Comparative Assessment Alternative Disposal Concepts	6, 9, 10 12, 14 15, 16, 17 19 26, 28, 32 41 61, 62, 63, 72, 73, 74, 75, 78, 82, 87, 88, 89, 96, 109 128, 132, 133, 134 180, 183, 186 188, 189 200, 203 206, 208, 211, 212, 214, 218, 219, 222, 224, 226, 229, 230, 233, 238, 242, 245, 247, 252, 257, 258 271, 272 283, 284 301, 302, 304 315, 317, 320, 322, 323, 324, 325, 328, 333, 338, 344, 346, 352, 363, 369, 370, 371, 376, 382, 383, 384, 390, 393, 399
219	Organization and Presentation Consequence Analysis Waste Management Operations Geologic Considerations Multibarriers for Disposal Socioeconomic/Sociopolitical Issues Comparative Assessment Alternative Disposal Concepts	28, 31 62, 108, 110 172 268 281, 282 293 311 315

APPENDIX C

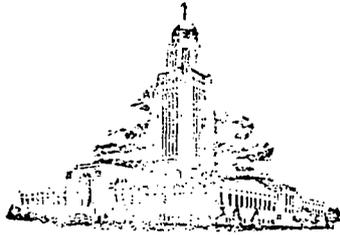
FEDERAL AND STATE COMMENT LETTERS

APPENDIX C

STATE AND FEDERAL AGENCY COMMENT LETTERS

This appendix contains reproductions of those letters submitted by federal agencies (Department of Commerce, Department of Health, Education, and Welfare, Department of Interior, Environmental Protection Agency, Federal Energy Regulatory Commission, and Nuclear Regulatory Commission) and cover letters of those comments submitted by state governments.

TELEPHONE
1-800-441-1000



STATE of NEBRASKA

STATE OFFICE OF PLANNING & PROGRAMMING
BOX 94601, STATE CAPITOL, ROOM 1321
LINCOLN, NEBRASKA 68509
(402) 471-2414

June 21, 1979

Dr. Colin A. Heath
Division of Waste Isolation
Mail Stop B-107
Department of Energy
Washington, D. C. 20545

Dear Dr. Heath

Under the provisions of OMB Circular A-95, Part I, this agency has completed a state level review of the draft environmental impact statement for Management of Commercially-Generated Radioactive Waste (DOE/EIS-0046-D).

The enclosed comments were received from the State Department of Health and the State Department of Environmental Control and should be considered by the Department of Energy in the development of the final statement.

Sincerely

Terry Kubicek
Natural Resources Coordinator

TK:mp
Enclosures
cc: Richard Beck
Dan Drain

CON STENBERG
DIRECTOR

North Carolina
Department of Administration
116 West Jones Street Raleigh 27611

James B. Hunt, Jr., Governor
Joseph W. Grimsley, Secretary

Division of State Budget and Management
John A. Williams, Jr., State Budget Officer
(919) 733-7061

July 13, 1979

MEMORANDUM

TO: Dr. Colin A. Heath, U.S. Dept. of Energy
FROM: Chrys Baggett, Director, State Clearinghouse
SUBJECT: File #145-79; Draft EIS - Management of Commercially
Generated Radioactive Waste

Attached are additional comments which were submitted following our clearance letter on your

- _____ Notification To Clearinghouse Of Intent To Apply For Assistance
- _____ Application For Funding
- _____ XXXXX Environmental Impact Statement
- _____ Environmental Review

If you have questions regarding these comments please contact me at (919) 733-7061

mw

Attachment(s)

C 2



State Of Idaho
DIVISION OF BUDGET, POLICY PLANNING AND COORDINATION
 EXECUTIVE OFFICE OF THE GOVERNOR

WIN V. EVANS
 Governor

STATE CLEARINGHOUSE
 JULY 3 1979

Statehouse
 Boise, Idaho 83720

WILLIAM P. CLEMENTS, JR.
 GOVERNOR



OFFICE OF THE GOVERNOR

July 3, 1979

DR COLIN HEATH
 DIVISION OF WASTE ISOLATION M/S B-107
 US DEPT ENERGY
 WASHINGTON, D.C. 20545

DR HEATH,

THE IDAHO STATE CLEARINGHOUSE HAS COMPLETED REVIEW OF THE US DEPT OF ENERGY'S
 DEIS, MANAGEMENT OF COMMERCIALY GENERATED RADIOACTIVE WASTE (OUR SAI #00692465)
 DOE/EIS-0046-D. THE FOLLOWING STATE AGENCIES AND AREAWIDE CLEARINGHOUSES RE-
 CEIVED COPIES OF THE DRAFT AND WERE ASKED TO SUBMIT COMMENTS:

- PANHANDLE AREA COUNCIL (COEUR D'ALENE)
- CLEARWATER ECONOMIC DEVELOPMENT ASSOCIATION (MOSCOW)
- ADA PLANNING ASSOCIATION (BOISE)
- REGION IV DEVELOPMENT ASSOC (TWIN FALLS)
- SOUTHEAST IDAHO COUNCIL OF GOVERNMENTS (POCATELLO)
- IDAHO OFFICE OF ENERGY
- NATURAL RESOURCES BUREAU (DIVISION OF BUDGET, POLICY PLANNING & COORDINATION)

THE AREAWIDE CLEARINGHOUSE, CLEARWATER ECONOMIC DEVELOPMENT ASSOCIATION HAS
 FORWARDED COMMENTS FROM THE NEZ PERCE TRIBE OF IDAHO. TRIBAL PLANNER,
 MICHAEL PENNEY HAS EXPRESSED CONCERN FOR THE POTENTIAL HAZARDS OF STORING &
 TRANSPORTING RADIOACTIVE WASTES. THE TRIBE FINDS THE POTENTIAL DAMAGE POSSIBLE
 FROM STORAGE AS PARTICULARLY HARMFUL TO HUMAN AND WILDLIFE POPULATIONS. IN SUM-
 MARY, "THE NEZ PERCE TRIBE OF IDAHO IS VERY CONSCIOUS OF THE IMPACT ON THE EN-
 VIRONMENT". THE STATE CLEARINGHOUSE STRONGLY URGES YOU TO ADDRESS THIS ISSUE
 OF CONCERN TO THE NEZ PERCE. PLEASE FORWARD SUFFICIENT COPIES OF THE FINAL
 STATEMENT TO THE STATE CLEARINGHOUSE WHEN THE PUBLICATION IS AVAILABLE. THANK
 YOU FOR PROVIDING US WITH COPIES OF THE DRAFT.

SINCERELY,

 PAM DEMIO-RYBUS
 COORDINATOR

cc: CEDA; M. PENNEY
 NEZ PERCE TRIBE
 EQUAL OPPORTUNITY EMPLOYER

0908

Dr. Colin A. Heath
 Division of Waste Isolation
 Mail Stop B-107
 U.S. Department of Energy
 Washington, D.C. 20545

Dear Dr. Heath:

The draft environmental impact statement entitled "Management
 of Commercially Generated Radioactive Waste", has been reviewed
 by the Budget and Planning Office and interested State agencies.
 Agency comments are enclosed for your information and use.

The Budget and Planning Office appreciates the opportunity to
 review this document. If we can be of any further assistance
 during the application process, please do not hesitate to call.

Sincerely,

Donald E. Harley, Manager
 Economic and Natural Resources
 Budget and Planning Office

DEH:jl

- Enclosures: Comments by -
- Railroad Commission
 - Parks and Wildlife Department
 - State Department of Highways and Public Transportation
 - Air Control Board
 - State Soil and Water Conservation Board
 - Department of Water Resources
 - Texas Department of Community Affairs

OFFICE OF THE GOVERNOR
STATE A-95 CLEARINGHOUSE



STATE OF VERMONT
MONTPELIER, VERMONT 05602

STATE PLANNING OFFICE
AREA CODE 802-828-3326



STATE OF NEVADA
GOVERNOR'S OFFICE OF PLANNING COORDINATION
CAPITOL COMPLEX
CARSON CITY, NEVADA 89710
(702) 865-4663

July 5, 1979

MEMORANDUM

To: Dr. Colin A. Heath, Division of Waste Isolation
Mail Stop B-107, U. S. Dept. of Energy
Washington, D. C. 20545

From: Emily Neary, A-95 Coordinator *en*

Date: July 3, 1979

Re: Dept. of Energy, Draft Environmental Impact Statement
"Management of Commercially Generated Radioactive Waste" - 2 volumes

As the State Clearinghouse under OMB Circular A-95 we have notified other public agencies with a possible interest in your: draft environmental impact statement.

Copies of comments received are attached: from the Division for Historic Preservation.

:enclosure

Dr. Colin A. Heath
Division of Waste Isolation
Mail Stop B-107
U.S. Dept Energy
Washington D.C. 20545

RE: SAI NV # 79300067 Project: DOE/EIS 0046-D
79300068 DOE/EIS 0026-D

Dear Dr. Heath:

Attached are the comments from the following affected State Agencies: Division of Environmental Protection, and Dept. of Energy concerning the above referenced projects.

These comments constitute the State Clearing house review of this proposal. Please address these comments in the final or summary report.

Sincerely,

Mike Nolan for
Robert M. Hill
State Planning Coordinator

RMH:md

Enclosures

C.4

STATE OF KANSAS

Department of  Administration

DIVISION OF STATE PLANNING AND RESEARCH

4th Floor—Mills Building
109 W. 9th
Topeka, Kansas 66612

July 10, 1979

Dr. Colin A. Heath
Div. of Waste Isolation
Mail Stop B-107
U. S. Department of Energy
Washington, D. C. 20545

Re: DOE/GEIS—Mgmt Of Commercially
Generated Radioactive Waste.
No. 7207

The referenced project has been processed by the Division of State Planning and Research under its clearinghouse responsibilities described in Circular A-95.

The Management of Commercially Generated Radioactive Wastes presents many special problems, not all of which have been addressed in this Draft GEIS, particularly since this draft statement is not site specific. This condition makes it difficult to determine which program strategies, if any, would be most appropriate for Kansas. Our review agencies have reviewed this statement in a general nature. If, for example, a specific strategy for managing radioactive waste for a future plant in Kansas (possibly Wolf Creek) was addressed, our agencies would have made more specific comments.

I have included agency comments for your information. I hope you will address our expressed concerns in the final environmental impact statement in the spirit and intent of the National Environmental Policy Act of 1969.

Should you have any questions please contact this office. Please refer to the State Application Identifier (SAI) Number above in all future correspondence.

Sincerely,


Paul V. DeGaeta
A-95 Coordinator

PVD:jc

Attachment

cc: Ruth Clusen, Ass't Sec. for Environment



TENNESSEE

STATE PLANNING OFFICE

660 CAPITOL HILL BUILDING
301 SEVENTH AVENUE, NORTH
NASHVILLE, TENNESSEE 37219
615-741-1676

July 6, 1979

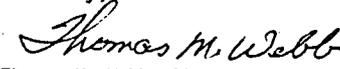

Dr. Colin A. Heath
Division of Waste Isolation
Mail Stop B-107
U. S. Department of Energy
Washington, D. C. 20545

Dear Dr. Heath:

Enclosed please find comments from the Tennessee Historical Preservation Officer and the Department of Public Health.

Your consideration of these comments will be appreciated.

Sincerely,


Thomas M. Webb, Director
Natural Resources Planning

TMW:brs

Enclosure

C. 5



STATE OF NEW JERSEY
 DEPARTMENT OF ENVIRONMENTAL PROTECTION
 DANIEL J. O'HEARN, COMMISSIONER
 P. O. BOX 1390
 TRENTON, N.J. 08625
 609-292-2865

June 29, 1979

Dr. Collin A. Heath
 Division of Waste Isolation
 Mail Stop B-107
 U.S. Department of Energy
 Washington, D.C. 20545

Dear Dr. Heath:

The State of New Jersey has reviewed DOE/EIS-0046-D, the Draft Environmental Impact Statement: Management of Commercially Generated Radioactive Waste. The sheer volume of the report is indicative of the thorough examination of the issue of radioactive waste disposal which is being pursued by federal agencies, especially the Department of Energy. Careful examination of the volumes shows the thorough study which has gone into this issue and the painstaking detail with which the nuclear fuel cycle and its relation to the generation of radioactive waste have been studied. The various forms of waste (glass, ceramics, etc.) are also reviewed in extensive detail.

And yet, what is the subject of the report? It is essential that a hazardous substance; namely radioactive waste, be isolated from the environment. Perhaps there are ten methods available, as described in the present document, or by a slightly different classification system, six, as described in the Interagency Review Group (IRG) Reports, the State's review of which was transmitted December 4, 1978. Many of the comments which applied to the IRG Reports apply to the present volumes.

We would recommend that the errata sheet which replaces page 1.3 be disposed of and the original page reinstated. In particular, the original words:

"The first disposal facilities for HLW will be mined repositories. Several geologic environments possessing a wide variety of emplacement media should be examined."

Dr. Heath

-2-

June 29, 1979

are superior to the new language:

"...near-term program activities should be predicated on the tentative assumption made for interim planning purposes that the first disposal facilities will be mined repositories."

The kindest thing that can be said about the second statement is that it is weak and indecisive. In any case, it is not representative of the positive leadership this country needs in resolving the issue of radioactive waste disposal.

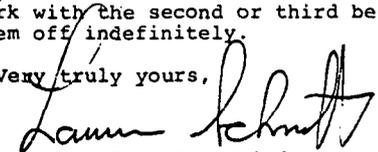
It is necessary to continue R&D in other means of radioactive waste disposal as an alternative of mined repositories develop more problems than their competitors. Eventually, a superior method may be found. In the meantime, it is essential to develop an operational facility. The discussion in 1.1.1 shows a much more pragmatic approach. It is stated that the first thousand years of disposal are most critical and that, based on our own knowledge of the languages of earlier investigations, it is reasonable to expect that the inhabitants of earth will recognize repository markers for millenia.

The relative toxicity of plutonium and lead, discussed on page 3.1.65 should be more widely promulgated. Although it does not mean that concern for plutonium be reduced, it does put the problem in better perspective.

On page 3.1.75, it is stated that "The majority of nuclear wastes are residuals from defense programs." Is this a measurement of volume, which could be reduced by evaporation, or is the measurement based on curies of activity?

This report, like the many which have preceded it, indicates that a large body of knowledge of how to treat radioactive wastes exists now. All of the problems haven't been solved but it is unlikely that much more will be learned until large scale pilot projects are begun. The position of the State of New Jersey is that now is the time to begin construction of such pilot projects, and as soon as feasible, a full-scale radioactive waste depository. It may be that the best type of facility and waste form will not be used but it would be best to begin work with the second or third best types now then to put this problem off indefinitely.

Very truly yours,


 Lawrence Schmidt, Chief
 Office of Environmental Review

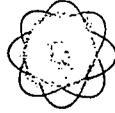
C.6



H. T. SUTTON
COMMISSIONER

State of Louisiana

OFFICE OF CONSERVATION



B. JIM PORTER
ADMINISTRATOR
NUCLEAR ENERGY DIVISION

-2-

July 9, 1979

Dr. Colin A. Heath
Division of Waste Isolation
Mail Stop B-107
U. S. Department of Energy
Washington, D.C. 20545

Dear Dr. Heath:

This is to acknowledge receipt of DOE/EIS-0046-D, "Management of Commercially Generated Radioactive Waste", Volumes I and II. We have reviewed the documents and found them to be very extensive and comprehensive in nature, covering all aspects of potential environmental consequences related to the disposal of high level and transuranic radioactive waste from the commercial nuclear fuel cycle.

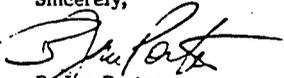
It is felt that since the United States will have to depend upon nuclear energy as a power source until other sources are feasible, it is imperative that there be a national program to ensure a solution to the problem of nuclear waste management and storage. While we agree that the approach to permanent disposal of nuclear waste should proceed on a step wise basis in a technically conservative manner, we also feel that additional delays such as have been experienced in the past, will have a detrimental affect on the establishment of a sound energy policy and on the nation's future energy outlook.

Basically, we agree with the analysis presented in the Draft GEIS by the DOE, that (1) the disposal of radioactive waste in geological formations likely can be developed and applied with a minimum environmental consequence; and (2) the program's emphasis, therefore, should be on the establishment of mined repository as the disposal technology for nuclear waste. As you know, the Louisiana Office of Conservation has been designated by Governor Edwards as the official liaison with DOE in the high level waste disposal investigations currently being conducted in and around Vacherie and Rayburn salt domes located in North Louisiana. We have in the past, and will continue to closely monitor the activities of the DOE, Division of Waste Isolation and its major contractor, the Batelle Memorial Institute, Office of Nuclear Waste Isolation in relation to ongoing studies in Louisiana, and will

provide comment on state concerns where appropriate. While the provisions of Act 650 of the 1978 Louisiana Legislature Regular Session do not allow the location of a storage facility within the State, we look forward to the development of a national strategy which addresses the technical and socio-political issues necessary for selection of repository sites and bringing repositories into operation within a reasonable period of time.

Thank you for the opportunity to comment on this vitally important issue.

Sincerely,


B. Jim Porter
Administrator
Nuclear Energy Division

BJP:pfid

C.7



WYOMING
EXECUTIVE DEPARTMENT
CHEYENNE

ED HERSCHLER
GOVERNOR

July 10, 1979

Dr. Colin A. Heath
Division of Waste Isolation
Mail Stop 13-107
U. S. Department of Energy
Washington, D.C. 20545

Re: Department of Energy's Draft
Environmental Impact State-
ment for the Management of
Commercially Generated
Radioactive Wastes

Dear Dr. Heath:

In compliance with the National Environmental Protection Act of 1969, Office of Management and Budget Circular A-95 (revised) and the Wyoming State Review Procedures, the State of Wyoming has completed its review of the subject draft environmental impact statement. Comments from the Wyoming Department of Environmental Quality are enclosed for your review and inclusion in the final statement.

Wyoming is not geologically suitable for the underground disposal of nuclear wastes. The salt and shale deposits within the state are located in the midst of valuable uranium, trona and petroleum deposits. The state's granite deposits are not only highly fractured and jointed but located near water tables.

The decisions made with regard to the isolation of nuclear wastes from the biosphere will be unprecedented in human history in terms of their lasting potential impacts on human health and the environment. Therefore, the uncertainties and risks associated with nuclear waste are of concern to every state.

The Department of Energy's desired goals of credibility and objectivity in the formulation of a nuclear waste strategy are hampered by the fact that the agency responsible for examining the feasibility of nuclear waste isolation is also responsible for the development of nuclear energy policies (and heretofore, the encouragement of the nuclear option).

Dr. Colin A. Heath
July 10, 1979
Page two

With this credibility handicap in mind, it is surprising that the Department of Energy's environmental conclusions and feasibility analysis appear to rely so heavily on convoluted logic and faulty, simplistic analogies. The fact that we run a greater risk of lethal contamination from the environmental presence of arsenic or cyanide than from high-level radioactive wastes will not assuage the public's concern over the safe disposal of these wastes (p.1.16). Similarly, the fact that some of the pyrimids are still standing and that we can decipher Egyptian hieroglyphics is no assurance that a nuclear waste depository will shield future generations from the hazards of strong doses of radiation (p. 1.6; 3.6364).

The final environmental statement should confront candidly the fundamental dilemma associated with nuclear waste management as well as implied policy objectives. The goals of nuclear waste management (i.e. the long-term protection of human health and the environment) cannot be ensured due to technical uncertainties. The feasibility of long-term isolation of nuclear wastes cannot be proved or disproved on the basis of experimentation, prototype testing or prior experience. Yet, the worst management alternative is to do nothing, allowing the radioactive wastes to remain exposed and saddling future generations with the burden of our neglect.

The Department of Energy, therefore, should make it clear that, due to our present state of technology, nuclear waste management will have to proceed on a step-by-step, trial-and-error basis. Permanent "disposal" of these wastes is not possible at this time.

The statement should also examine thoroughly the relationship between program strategy, environmental and health goals, and environmental/design standards. The draft statement fails entirely to examine this crucial relationship. The text of the draft implies that DOE expects standards to be set according to specific technical capabilities.

The final report should examine the feasibility and desirability of basing standards on health and environmental effects instead of on present technology. Even though standards based on health effects would have to consider the present state-of-the-art, the Department of Energy should consider health standards as being technology-forcing. Such

C. B

TWIN TOWERS OFFICE BUILDING
2600 BLAIR STONE ROAD
TALLAHASSEE, FLORIDA 32301



BOB GRAHAM
GOVERNOR
JACOB D. VARN
SECRETARY

STATE OF FLORIDA
DEPARTMENT OF ENVIRONMENTAL REGULATION

Dr. Colin A. Heath
July 10, 1979
Page three

an interpretation would buttress the necessity of a cautious step-by-step approach and the initial retrievability of the wastes.

I urge your careful consideration of these comments.

Sincerely,

Dick Hartman
State Planning Coordinator

DH/lđ

Attachment

July 27, 1979

Dr. Colin A. Heath
Division of Waste Isolation
Mail Stop B-107
Department of Energy
Washington, D.C. 20545

Dear Sr. Heath:

Please consider this public document. As a systems analyst, it expresses a number of my own concerns regarding the Draft EIS DOE/EIS-0046-D.

Sincerely,

J. S. Sherman

JSS/js

Enclosure

C. 9



LEE SHERMAN DREYFUS

STATE OF WISCONSIN
OFFICE OF THE GOVERNOR
STATE CAPITOL
MADISON, 53702

July 27, 1979

Telephone Number
(608) 266-1212

Dr. Colin A. Heath
Division of Waste Isolation
Mail Stop B-107
U. S. Department of Energy
Washington, D.C. 20545

Dear Dr. Heath:

Re: DOE/EIS-0046-D-Management of Commercially
Generated Radioactive Waste

The State of Wisconsin is aware of the sometimes conflicting, but urgent, issues related to the nuclear industry since we rely on nuclear power plants to provide 30 percent of our electrical energy.

While we recognize the primary Federal role in these issues, the problem of nuclear power and radioactive waste disposal are also state concerns and we will accept our responsibilities in these matters.

Wisconsin has a long history of accountable government involvement in proposals affecting the welfare of its citizens. I intend to maintain and improve this trust especially for nuclear waste disposal because of its serious implications to the energy and environmental future of Wisconsin and the Nation.

The responsibility over nuclear power and disposal of radioactive wastes must be a state and federal partnership. The Federal Government must make a special effort to recognize and comprehensively involve the states, local units of government and citizenry in all phases of the nuclear decision-making process.

The information contained in this Draft Environmental Impact Statement has serious overtones toward the future of our state, region and the Nation. The attendant problems will require our full and thorough attention. In order to begin a partnership approach of resolving these problems, I have directed several state agencies to provide my office with an interdisciplinary review of this Draft Environmental Impact Statement. These comments are attached.

Our review of the DEIS identified several serious inadequacies.

I feel the objectives to provide evidence supporting a specific program have not been substantiated by the information provided in this text.

I am confident that our comments will prove useful in preparing a final document which will be considered adequate within the spirit and intent of the National Environmental Policy Act, case law and the guidelines of the President's Council on Environmental Quality.

We are prepared to assist in any way possible to fulfill our obligations in this matter.

Sincerely,

Lee Sherman Dreyfus
GOVERNOR

Attach.

- cc: Honorable Jimmy Carter, President
- Harold R. Denton, Director, Nuclear Regulatory Commission
- Honorable Albert Quie, Governor of Minnesota
- Members of National Governors Association
- Douglas Costle - EPA, Washington
- John McGuire - EPA, Region V, Chicago
- Honorable Gaylord Nelson
- Honorable William Proxmire
- Members, Wisconsin State Legislature
- Stanley York - PSC
- Donald C. Percy - H&SS
- Robert Durkin - H&SS
- Lowell Jackson - DOT
- Mike Early - DLAD
- Ken Lindner - DOA
- M. E. Ostrom - Geo. & Natural History
- Honorable Bronson LaFollette - Attorney General
- John Stolzenberg - Leg. Council Office
- Anthony Earl - DNR

C.10



OFFICE OF THE GOVERNOR

WILLIAM P. CLEMENTS, JR.
GOVERNOR

July 31, 1979

Dr. Colin A. Heath
Division of Waste Isolation
Mail Stop B-107
U. S. Department of Energy
Washington, D. C. 20545

Dear Dr. Heath:

The Budget and Planning Office recently conducted a review of the draft environmental impact statement entitled "Management of Commercially Generated Radioactive Waste" prepared by your office.

Subsequent to the completion of that review, the enclosed comments from the Texas Department of Agriculture were not included with those of the other reviewing agencies. Those comments are being forwarded to you at this time for your use in the preparation of the final environmental impact statement.

I hope that the delay in the receipt of these comments will be of no inconvenience to you.

Sincerely,

Donald E. Harley, Manager
Economic and Natural Resources
Budget and Planning Office

DEH:jl

Enclosure



TENNESSEE DEPARTMENT OF CONSERVATION
TENNESSEE HISTORICAL COMMISSION
4721 TROUSDALE DRIVE, NASHVILLE 37220
615/741-2371

September 17, 1979

Dr. Colin A. Heath
Division of Waste Isolation
Mail Stop B-107
U.S. Department of Energy
Washington, D.C. 20545

Re: Draft Environmental Impact Statement-Management of Commercially Generated Radioactive Waste

Dear Dr. Heath:

The Tennessee Historical Commission, a state agency, has reviewed the above project in accordance with the provisions of OMB Circular A-95. Based on the information supplied, we conclude that the project as planned will not affect the plans or priorities of our office as a state review agency.

In addition to and separate from our A-95 review, the Executive Director, in his role as State Historic Preservation Officer, has also reviewed the undertaking with regard to National Historic Preservation Act compliance by the participating federal agency or its designated representative. Compliance procedures are set forth in the Advisory Council on Historic Preservation procedures for the protection of historic and cultural properties (36 CFR Part 800).

Our only recommendation at this time is that selection of specific waste treatment construction sites in our area should be coordinated with the State Historic Preservation Officer. Such coordination insures the proper identification of architectural, historic, archaeological and cultural properties included in or eligible for inclusion in the National Register of Historic Places as prescribed at 36 CFR 800.4 (see enclosure).

Please contact our office for assistance. Nick Fielder or Suzanne Whittenburg will be glad to address any questions or comments that you may have. Your cooperation is appreciated.

Sincerely,

Herbert L. Harper
Executive Director and
State Historic Preservation Officer

HLH:SDW:sh

xc: Thomas M. Webb, State Clearinghouse

C.11



ALBERT H. QUIE
GOVERNOR

STATE OF MINNESOTA

OFFICE OF THE GOVERNOR

ST. PAUL 55155

Mr. Jack Halzman
Page 2

Mr. Jack Halzman
Department of Energy
175 West Jackson Boulevard
11th Floor
Chicago, Illinois 60604

Dear Mr. Halzman:

RE: Draft Environmental Impact Statement:
Management of Commercially Generated Waste
DOE/EIS 0046 D
April, 1979
Public Hearing Comments
State of Minnesota
Chicago, Illinois
August 8, 1979

I appreciate this opportunity to present the State of Minnesota's position on the draft environmental impact statement for the management of commercially generated radioactive waste (DOE/EIS 0046 D) and request that this letter be made a part of the record of this public hearing.

The State of Minnesota has had a continuing interest in the affairs surrounding nuclear power. We have, in the past, participated in rulemaking and licensing proceedings. Correspondingly, Minnesota has an interest in the question of nuclear waste disposal. This interest is heightened by the interpretation of the Wisconsin Department of Natural Resources that equates one of the reference sites in the EIS to the specific geological makeup of an area in central Minnesota.

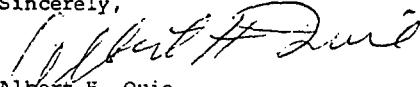
In 1977 the Minnesota Legislature enacted statutory restrictions on the importation of radioactive wastes into the State and the siting of a radioactive waste management facility in Minnesota (Minn. Stat. §116C 71-74). Although the statute was signed by my predecessor, Governor Perpich, it enjoys my full and complete support. I oppose the siting of a radioactive waste disposal facility in the State of Minnesota, except that which may be needed and feasible to store wastes from Minnesota's own nuclear generating plants.

The Draft EIS does a commendable job in identifying several potentially useful disposal strategies. The range of factors to be addressed is complete.

However, the EIS should present an analysis of the methods to be used by the Department of Energy in combining and weighing all factors in order to reach decisions on actual disposal sites for radioactive waste.

In sum, the State of Minnesota supports and encourages efforts by the federal government to address the problems of radioactive waste disposal. As Governor of Minnesota, I am opposed to the siting of a radioactive waste facility in this State except for wastes generated within our own State and only following thorough and intensive exploration at the national level of all alternatives available to solve this problem.

Sincerely,


Albert H. Quie
Governor of Minnesota

C.12





Office of the Governor

Atlanta, Georgia 30334

George Busbee
GOVERNOR

September 24, 1979

Ms. Ruth Clusen
Page 2
September 24, 1979

Ms. Ruth Clusen
Assistant Secretary for Environment
U. S. Department of Energy
Washington, D. C. 20585

Dear Ms. Clusen:

The State of Georgia appreciates the opportunity to review and comment on the Department of Energy's draft "Generic Environmental Impact Statement on the Management of Commercially-Generated Radioactive Waste," DOE/EIS-0046-D.

This Draft Environmental Impact Statement (DEIS) has serious implications upon the welfare of future generations and the quality of the environment in our State, the Region and the Nation. The stated strategy that "program emphasis should be on the establishment of mined repositories (in geologic formations) as the operative disposal technology" is of particular concern to the State of Georgia. It is appropriate to reiterate our position on this matter: We are unalterably opposed to those waste management operations which could result in the eventual contamination of groundwater sources. Although this DEIS acknowledges that groundwater "is a valuable and widely used resource" and "presumes that a repository should not affect the quality or availability to an unacceptable level," it concludes, however, that the migration of wastes into groundwater is potentially very high. Acceptance of any contamination of these resources would be irresponsible. The DEIS statement (p. 3.1.49) "Site selection will probably avoid areas of known major aquifers initially" should be responsibly changed to read: "Site selection must avoid any areas of aquifers." Our great natural groundwater resources are vital to the projected orderly growth and development of Georgia.

Our review of the DEIS identified several other serious inadequacies. These comments, compiled by Georgia's Environmental Protection Division, are attached.

Whereas, the State of Georgia is not opposed to the general concept of geologic disposal, the State is unalterably opposed to the specific concept relevant to a waste management strategy involving bedrock storage at the Savannah River Plant because of the potential contamination of the Tuscaloosa aquifer. It is quite apparent that DOE is proceeding to further develop a bedrock storage facility at the Savannah River Plant. Although the USEPA and the National Academy of Sciences have opposed this SRP waste disposal concept and the State of Georgia has stated its position on numerous occasions regarding this specific issue, it is the position of the State of Georgia that the Department of Energy has been neither sensitive nor responsive to Georgia's concerns in this matter. The State of Georgia will exercise all available options relative to this issue in order to protect its natural resources and the health and safety of the citizens of our State.

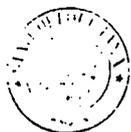
Sincerely,

George Busbee

GB:rpj

cc: Dr. C. A. Heath ✓
Division of Waste Isolation, USDOE

C.13



LOUISIANA HOUSE OF REPRESENTATIVES
COMMITTEE ON NATURAL RESOURCES

Box 44012, Capital Station
Baton Rouge, Louisiana 70804
Telephone: (504) 342-2350

October 3, 1979

COMMITTEE MEMBERS
REP. W. J. "BILLY" TAUZIN
CHAIRMAN
REP. J. CHRIS ULLO
VICE CHAIRMAN

REP. CLYDE F. BEL, JR.
REP. LEONARD J. CHABERT
REP. STEVEN J. DUPUIS
REP. MANUEL A. FERNANDEZ
REP. JESSE J. GUDRY
REP. TED M. HAIK, JR.
REP. MIKE L. HOGAN
REP. JOHN N. JOHN III
REP. CONWAY LOBLEU
REP. FRANK J. PATTI
REP. EDWARD C. SCOGIN
REP. JOHN W. SCOTT
REP. LAWRENCE A. SHERIDAN
REP. JOHN J. SIRACUSA
REP. ARTHUR W. SOUR

Louisiana's salt domes. It further prohibits the testing of geologic structures in Louisiana for their suitability for storage unless notice is given to the local parish governing body, the natural resources committees of the House and Senate of the legislature and the secretary of the Department of Natural Resources and anyone of those entities may cease the testing by objecting in writing. The test results must be made available to the House and Senate Natural Resources Committees upon completion of the testing.

The passage of this legislation was not intended to be vain and useless, but was intended to protect to the greatest extent possible the health and welfare of the citizens of our state and to prevent them from becoming guinea pigs. It was intended to assure that we are given the opportunity to question and probe any action the DOE may take to use Louisiana salt domes for the storage of radioactive waste. Whether or not this legislation is constitutional and enforceable, we feel that it will serve notice on those persons who make the final decision as to the site for any storage of waste, that Louisiana citizens must be a part of that decision making process in all respects.

Louisiana, has as a matter of state policy favored the development of nuclear energy and two nuclear power plants are presently being constructed within our state. We expect that those plants will some day provide electricity to our citizens and that those two plants will also generate waste which must be disposed of. We realize that on-site disposal will be available for only a few years and therefore, the natural resources committee recognizes the urgent need for permanent and safe disposal facilities. We support the idea of regional siting of repositories as opposed to one site or sites in all states with nuclear power plants. We realize that it would be impossible to require each state to handle its own waste and yet we feel that no one state should be required to be a dumping ground for the rest of the country.

We realize that a permanent facility will be needed within a few years

COMMITTEE STAFF
KATHLEEN R. WOODSON
COUNSEL
KATHLEEN R. WOODSON
SECRETARY

Division of Waste Isolation (ET-960)
Mail Station B-107
U.S. Department of Energy
Washington, D. C., 20545

RE: Statement on the draft Environmental Impact Statement
Management of Commercially Generated Radioactive Waste

We appreciate this opportunity to comment on the environmental impact statement which we have received on the management of commercially generated radioactive waste. It is apparent that the Department of Energy is now moving in the direction of making decisions that will be of enormous concern to the people of this country involving the permanent disposal of commercially generated radioactive waste. In no place does that issue concern the average citizen more than it does in Louisiana. The citizens of our state are raising this issue during this election in Louisiana in a manner that impresses upon elected officials the concern that our citizens are feeling.

As elected officials, it is our responsibility to represent the wishes of our constituency and we also feel that because we are given opportunities to obtain information not available in all instances to the average citizen, it is our responsibility to provide leadership and to take a position on controversial and confusing issues.

Therefore, the Louisiana Legislature has taken a strong and forceful stand on the issue of disposal of radioactive waste and that stand is incorporated in Act 650 of 1978 which prohibits the disposal of radioactive waste in

as storage facilities begin to reach their capacity. We would urge you to seek input from local and state governments as well as ordinary citizens in a more aggressive manner by holding hearings in the specific areas that are being considered and by seeking out the advice of a broad range of local and state officials. Holding hearings in Atlanta, Washington, New York, or Dallas just simply does not allow input by the people most affected. We hope that you will then reflect on the advice you will receive and give equal weight to the advice of state officials as well as federal.

Regardless of the type of storage medium or technology used, the most difficult issue to resolve will be the siting of each facility. Although this EIS does not address specifically each site under consideration, we have seen the ONWI summary characterization and recommendation of study areas for the Gulf Interior Region issued in May, 1979, and we realize that there are two salt domes in Louisiana under consideration by DOE.

Although we can't claim expertise in the various scientific disciplines which are discussed in the EIS, we do have expertise in the assessment of Louisiana's political, economic, and psychological environment. In considering the site of a repository, more must be considered than the geologic or surface utilization of the area involved. The decision makers must consider the political, economic, and psychological effects of their actions. We would urge that when specific sites are under consideration that informed local and state officials must be a part of the process of choosing a site or of determining that a site is inappropriate. In the past, most decisions made by DOE and other federal agencies involving our state have not allowed sufficient local input and that is probably one of the main reasons that many of these programs have run into trouble. A prime example is the Strategic Petroleum Reserve Program in which the DOE has taken several of our salt domes and converted them to oil storage facilities. In their haste to get this program under way, they have ignored citizens and their elected representatives and have ignored concern

for our fragile ecology.

In addition, Louisiana has experienced a number of environmental problems in the past several years. The ONWI report ignores some of the most important environmental and economic issues concerning our citizens. It makes no mention of the fact that Louisiana has been the dumping ground of the nation for hazardous waste. Louisiana is the first state in the nation to implement a strict hazardous waste permitting program. During the past year, we discovered that there are numerous illegal and in some cases abandoned chemical waste dumps across the state. The clean up of the abandoned facilities will be a financial burden for our state. It is expected that we will have to deal with this environmental problem for quite some time before we finally get it under control.

Our waterways are experiencing the stress of our economic growth and of our agricultural production. In north Louisiana we have had serious problems with pesticide contamination of our lakes, rivers and streams. In both north and south Louisiana we have areas that are non-attainment for certain air pollutants.

In north Louisiana, we have recently discovered that we have millions of tons of lignite coal deposits which we will soon be mining. We will undertake a program to allow surface mining along with reclamation programs to follow so that we can make this coal available as a fuel for our industries. The production of coal will mean greater industrial development and expansion all across that area of our state with both power plants and other support facilities expected to be constructed close to the mining area.

We are a state of approximately 4,000,000 people living on 48,506 square miles of land and water. This means there are more than 81.8 people per square mile. In Webster Parish where the Vacherie dome is located there are 42,068 people living on 620 square miles or 67.9 people per square mile. In Bienville

Parish where the Rayburn dome is located there are 17,226 people living on 858 square miles or 20.1 people per square mile. Those areas are populated and at present are primarily agricultural. The people of these areas have historically been poor people and in particular the minority groups have suffered economic deprivation. Louisiana has moved into a period of prosperity for all our citizens because of our status as an energy producer for the rest of the country. With the aggressive mining of coal, reclamation, and the industry which will develop in the coal producing area, these people will be able to look forward to a prosperity that they have never known. A radioactive waste repository will probably put an end to their dreams of prosperity and to their expectations of a better way of life.

With all the environmental problems that I have mentioned, it is clear that Louisiana cannot accept the additional environmental burden of storing radioactive waste in our salt domes within the near future. We have willingly accepted the environmental problems of producing energy for our nation and we feel that our citizens have done more than their share in the area of shouldering the energy needs of the rest of the country.

We are further concerned that our salt domes are an important natural resource and may be useful for many better purposes than storage of radioactive waste which use would eliminate all alternate uses known and yet to be discovered. We would encourage additional research as to the nature and alternative uses of salt domes. We would not want to foreclose other alternative uses which might be of greater benefit to our citizens.

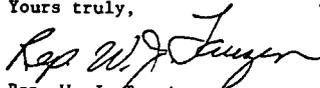
Regardless of the outcome of any EIS, as to the environmental suitability of an area, the most important environmental factor to be considered is the acceptance of the people living in the area. Without that acceptance, no facility will possibly be safe and useful. Whether the facility is harmful to the bodies of our citizens or not, we must be assured that it is not

harmful to their minds as well and that they will be able to lead normal and anxiety free lives. We believe that wherever the repository is located, compensation to both the residents and the governments of the area should be made to reduce the economic and environmental burdens imposed.

The Department of Energy signed a letter of agreement last year with our governor, agreeing not to use our salt domes for radioactive waste disposal without the states permission. We would hope that the department under this and any new administration will continue to honor that agreement.

Thank you for this opportunity to make these comments which I hope will give you some guidance as you prepare to make decisions regarding the first of the repositories for storage of radioactive waste.

Yours truly,


Rep. W. J. Fauzin
Chairman, House Natural Resources Committee

C. Frank Marscher, III
Secretary



COMMONWEALTH OF KENTUCKY
DEPARTMENT FOR NATURAL RESOURCES AND ENVIRONMENTAL PROTECTION
OFFICE OF THE SECRETARY
OFFICE OF POLICY AND PROGRAM ANALYSIS
CAPITAL PLAZA TOWER
FRANKFORT, KENTUCKY 40601
PHONE (502) 564-7320

October 2, 1979

Julian M. Carroll
Governor



TENNESSEE

STATE PLANNING OFFICE

660 CAPITOL HILL BUILDING
301 SEVENTH AVENUE, NORTH
NASHVILLE, TENNESSEE 37219
615-741-1676

October 4, 1979

Department of Energy
Division of Waste Isolation
M.S. B-107
Washington, D. C. 20545

RE: Management of Commercially Generated Radioactive Waste

Dear Sirs:

The Draft Environmental Impact Statement prepared on the Management of Commercially Generated Radioactive Waste has been circulated to the Kentucky Environmental Review Agencies for their comments. Attached are the comments that have been returned by them. Any late arriving comments will be forwarded to your attention.

Sincerely,

Boyce R. Wells
Environmental Review Coordinator

BRW:bsc

Dr. Colin A. Heath
Division of Waste Isolation (ET-960)
U. S. Department of Energy
Mail Station B-107
Washington, D.C. 20545

Dear Dr. Heath:

Enclosed are comments regarding the EIS for the Management of Commercially Generated Radioactive Waste.

If this office can be of further assistance please do not hesitate to contact me.

Sincerely,

Saralee W. Terry
Resource Analyst

SWT:pbw

Enclosure

C:17



M. PETER LANAHAN, JR.
FIRST DEPUTY COMMISSIONER

STATE OF NEW YORK
DEPARTMENT OF
ENVIRONMENTAL CONSERVATION
ALBANY, NEW YORK 12233

FEDERAL ENERGY REGULATORY COMMISSION
REGIONAL OFFICE
730 Peachtree Street, N. E.
Atlanta, Georgia 30308
June 20, 1979

U.S. Dept. of Energy
Attn: Div. of Waste Isolation
MSB-107
Washington, D.C. 20545

Gentlemen:

The State of New York has completed its review of the draft Generic Environmental Impact Statement (GEIS) "Management of Commercially Generated Radioactive Waste" (DOE/EIS-0046-D), issued April 1979.

We must commend the DOE staff on its objectivity and thoroughness in the preparation of this document. We concur with the DOE position that conventional geological disposal is the option which is best understood and provides the best means for monitoring and control of the waste. Due to the urgent need for a national waste repository, this should be the option of first choice.

Thank you for the opportunity to comment on the GEIS. We hope our attached comments will be useful in the preparation of your final report.

Sincerely,

M. Peter Lanahan, Jr.
First Deputy Commissioner

attachment

Dr. Carlin A. Heath
Division of Waste Isolation
Mail Stop B-107
U. S. Department of Energy
Washington, D. C. 20545

Dear Dr. Heath:

This refers to the Draft Generic Environmental Impact Statement (GEIS) on management of commercially generated radioactive waste, received by the Atlanta Regional Office on May 30, 1979.

The Commission's principal area of responsibility is the regulation of the electric power and natural gas industries. Therefore, it is concerned with the possible impact of radioactive waste disposed on the construction and operation of bulk electric power and natural gas facilities. Since at the present there are commercially operated nuclear generating units interconnected into the power system, and other units under construction, we too are concerned with the ultimate disposition of radioactive wastes.

In reviewing the study, we note that there are potential geologic burial areas located throughout the contiguous United States as illustrated in Figures 3.1.1, 3.1.2, 3.1.3, and 3.1.4. Since the FERC has licensed hydroelectric generating facilities located throughout the U.S., of particular concern to us are the problems resulting from both routine operations and accidental spills over watershed areas upstream of FERC licensed hydroelectric projects that may affect generation. In addition, when selecting geologic waste disposal areas, attention should be given to the effects upon potential hydro sites which may be, or become, economically feasible to develop.

We appreciate the opportunity to comment on the GEIS.

Very truly yours,

Arne O. Kauranen
Regional Engineer

C.18



UNITED STATES DEPARTMENT OF COMMERCE
 The Assistant Secretary for Science and Technology
 Washington, D.C. 20230
 (202) 377-3333 4335

July 6, 1979

Dr. Colin A. Heath
 Division of Waste Isolation
 Mail Stop B-107
 U.S. Department of Energy
 Washington, D.C. 20545

Dear Dr. Heath:

This is in reference to your draft environmental impact statement entitled, "Management of Commercially Generated Radioactive Waste." The enclosed comments from the National Oceanic and Atmospheric Administration and the Maritime Administration are forwarded for your consideration.

Thank you for giving us an opportunity to provide these comments, which we hope will be of assistance to you. We would appreciate receiving ten (10) copies of the final statement.

Sincerely,

Sidney R. Gayler
 Sidney R. Gayler

Enclosures from: Mr. Gordon Lill
 National Ocean Survey
 NOAA

Mr. F. Chew
 Environmental Research Laboratories
 NOAA

Mr. Douglas M. LeComte
 Environmental Data and Information
 Service
 NOAA

Capt. George Steinman
 Maritime Administration



UNITED STATES DEPARTMENT OF COMMERCE
 National Oceanic and Atmospheric Administration
 NATIONAL OCEAN SURVEY
 Rockville, Md. 20852

0A/C52x4:CWC

TO: PP - Richard L. Lehman

FROM: *Gordon Lill*
 for OA/Cx1 - Gordon Lill

SUBJECT: DEIS #7904.32 - Management of Commercially Generated
 Radioactive Waste

The subject statement has been reviewed within the areas of NOS responsibility and expertise, and in terms of the impact of the proposed action on NOS activities and projects.

The following comments on Section 3.6, The Sub-seabed Geologic Disposal Concept, are offered for your consideration.

More emphasis should be placed on understanding the nature of transport of materials in the water column in areas under study for waste disposal. Ongoing programs in quantifying biological pathways should be expanded and a comprehensive program of physical, chemical and biological measurements should be undertaken and models developed for deep ocean layers. It is essential to quantify what will happen in the case of accidental breakage or unexpected leakage in terms of the water column serving as an emergency barrier.

C-19



U.S. DEPARTMENT OF COMMERCE
National Oceanic and Atmospheric Administration
ENVIRONMENTAL RESEARCH LABORATORIES

AOML/PhOL
15 Rickenbacker Causeway
Miami, Florida 33149

- 2 -

Date: May 25, 1979

To : R. Lehman, PP/EC
Office of the Administrator, NOAA, Washington, D.C.

From: F. Chew, AOML/NOAA *Frank Chew*

Subj: Comments and suggestions on the physical oceanographic aspects of
DEIS 7904.32 - Management of Commercially Generated Radioactive
Waste

As instructed by Dr. D. Hansen, Director (Acting), AOML, this is in reply
to your memo of April 26, 1979.

A. Island Disposal

1. Page 3.5.21, line 13 from top:

The hydraulic gradient and the stability of freshwater lens may also
be affected by sea level slope induced by coastal currents and topo-
graphic focusing of surface and internal gravity waves.

2. Page 3.5.23, line 9 from bottom:

Sea level slope is another factor in the stability of freshwater lens.

3. Page 3.5.24, line 8 from top:

Factors of importance to sediment patterns and movements include tides,
land discharges, coastal circulation, littoral currents induced by
surface gravity waves, and direct wave actions during storm and Tsumani.

4. Page 3.5.25, line 13 from top:

Replace the sentence beginning with "However", on line 13 with:
However, their natures and time variant behaviors, particularly of
subsurface currents, are incompletely understood.

5. Page 3.5.25, line 15 from bottom:

Replace the sentence beginning with "Similar" on line 15 with:
For oceanographic data there is need for both intensive and extensive
observations over a long period in order to resolve the time and
space scales of the coastal, littoral, and other flows.

8. Sub-seabed Disposal

1. Page 3.6.1, line 23 from top and page 3.6.4, line 18 from bottom:

A surface current gyre is a partially closed circulatory (not circular)
system of surface and upper layer waters.

2. Page 3.6.6, line 6 from bottom (including footnote):

Supposing the correctness of the assumptions, the breakthrough time,
T, is not a million years, but is

$$T = D^2/A = (100 \text{ m})^2 / (3 \times 10^{-6} \text{ cm}^2/\text{yr}) = 3 \times 10^{13} \text{ yrs.}$$

3. Page 3.6.7, line 19 from bottom:

The bottom flow is not uniform and plate-like. The mean flow is
thought to be slow, but there is a transient component that is
related principally to the tides. Within about 5 m of the bottom
there is a boundary layer that is thought dynamically analogous to
the lowest 2 km in the atmosphere. See Wimbush, M. and W. Munk
(1970): The Benthic Layer in THE SEA, Vol. 4, pt. 1, A. E. Maxwell,
Editor.

4. Page 3.6.7, line 17 from bottom:

Delete estimates of barrier properties, or add "if any" to read:
to allow estimates of the barrier properties, if any, of the water
column,....

5. Page 3.6.17, line 1 from top:

Replace sentence beginning on line 1 with: The long-term stability
of the patterns suggests a weak current regime.

6. Page 3.6.20, line 10 from bottom:

Replace "a poor" with "no" in the sentence beginning on line 10 so
so that it reads: For reasons given, the water column is likely to
be no barrier against large quantities of nuclides....

7. Page 3.6.30, line 16 from top:

Replace "deep physical circulation and biological processes" with
all relevant processes. Delete footnote on same page.



C-20



UNITED STATES DEPARTMENT OF COMMERCE
National Oceanic and Atmospheric Administration
ENVIRONMENTAL DATA AND INFORMATION SERVICE
Washington, D.C. 20235

June 21, 1979

OA/Dx61

TO: PP/EC - R. Lehman
FROM: OA/Dx61 - Douglas M. LeComte *Douglas M. LeComte*
SUBJECT: DEIS 7904.32 - Management of Commercially Generated
Radioactive Waste

- 3 -

C. Icesheet Diaposa

This part is adequate.

cc: Dr. D. V. Hansen

Page 3.7.21, Global and Polar Climatology: Mention should be made of recent research which shows that continued use of fossil fuels for energy production could increase atmospheric carbon dioxide levels to the extent that the resulting enhanced "greenhouse effect" would significantly raise global air temperatures. Studies of nuclear waste disposal in polar ice should consider that global circulation models which assume increased CO-2 levels indicate that the largest warming trends would be in Polar regions. An increase in annual mean temperatures by 12°C or more well before the end of the next century is possible, according to these models. A climate change of such a large magnitude would likely cause significant changes in the depth of Polar ice and snow, especially in the Northern Hemisphere.

C.21





UNITED STATES DEPARTMENT OF COMMERCE
Maritime Administration



UNITED STATES ENVIRONMENTAL PROTECTION AGENCY
WASHINGTON, D.C. 20460

SEP 27 1979

May 30, 1979

MEMORANDUM FOR: Dr. Sidney R. Galler
Deputy Assistant Secretary for Environmental Affairs

Subject: U.S. Department of Energy - Draft Environmental Impact
Statement (DEIS) concerning Management of Commercially
Generated Radioactive Waste (April 1979)

Dr. Colin A. Heath, Director
Division of Waste Isolation, MS B-107
Department of Energy
Washington, D.C. 20545

Dear Dr. Heath:

The Environmental Protection Agency (EPA) has reviewed the Department of Energy's (DOE) Draft Environmental Impact Statement (DEIS) for "Management of Commercially Generated Radioactive Waste," (DOE/EIS-0046-D). Our detailed comments are enclosed.

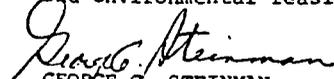
There are a number of serious deficiencies in the analysis which should be corrected in the Final EIS. They are: (1) failure to consider the time-integrated population dose as an important parameter in evaluating the impact from the waste disposal; (2) failure to consider individual dose to ground water users; (3) acceptance of a level of exposure comparable to background radiation (identified in the DEIS as 120 millirem per year) as a permissible additional dose to individuals; (4) lack of a sensitivity analysis showing which parameters in the risk analysis are important; (5) use of outdated, questionable, and/or one-sided radiobiology references; (6) occasional improper consideration of waste chemistry and geochemistry; (7) incomplete economic analysis; and (8) failure to relate radiation doses to health effects. Also, some information is lacking, making a good comparison of options for disposal of radioactive waste incomplete.

It is to be noted that "Sub-seabed disposal" would be subject to the dumping requirements of the Marine Protection, Research, and Sanctuaries Act of 1972 and that the dumping of high-level waste is prohibited by the Act. The Final EIS should reflect these facts.

In addition, in preparing the Final EIS, reference is needed to the present development by EPA of Federal guides for radioactive waste management and standards for high-level radioactive waste. DOE should consider the requirements stated in the proposed criteria published in 1978 (43 F.R. 53262 et seq.). The proposed criteria are under review

The subject generic DEIS has been reviewed as requested by your memorandum of April 24, 1979. This Statement examines ten alternative methods for disposal of nuclear wastes and evaluates their anticipated environmental impacts. Based on the analysis presented, the Department of Energy (DOE) proposes that (1) the disposal of radioactive wastes can likely be developed and applied with minimal environmental consequences and (2) the program emphasis should be on the establishment of mined repositories as the operative disposal technology. DOE further recommends that R&D on technical operations other than mined repositories should be performed for the nearer-term approaches (i.e., deep ocean sediments and very deep holes) so that they may be adequately evaluated as potential competitors.

Our comments address the disposal of high level radioactive wastes by deep ocean emplacement. Although this at-sea disposal method is currently prohibited by international treaty and national law, this disposal option should continue to be evaluated, but on a priority basis, for future use. The sub-seabed geologic disposal concept using the central regions of sediment-covered subocean tectonic plates offers potential advantages, not the least of which are tectonic stability, environmental stability, absence of resources, and remoteness from man's activities. A special element of this disposal option is the development of a safe transport/emplacement ship system. Preliminary design and safety analyses of this transport/emplacement ship system, as well as the related port facility, should be initiated to parallel other technical and environmental feasibility studies.


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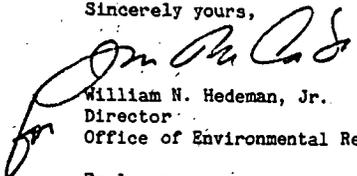
for promulgation as Federal guides and are scheduled to be published before the end of 1979. We will also publish our high-level waste standards in draft form in several months. These general guides and specific standards will identify what must be accomplished in waste management activities to provide assurance of public health protection and environmental preservation.

With proper attention to the above concerns, we believe the Final EIS can support a continuing program to develop a safe disposal system for high-level radioactive waste.

However, because we have reservations concerning the environmental effects of certain aspects of the proposed program, we rate this Draft EIS ER-2. This rating will be published in the Federal Register.

Should DOE have questions about our comments, please call Betty Jankus (NEPA matters - 755-0770) of my staff.

Sincerely yours,



William N. Hedeman, Jr.
Director
Office of Environmental Review (A-104)

Enclosure

Detailed Comments of the
United States Environmental Protection Agency
on the
United States Department of Energy's
Draft Environmental Impact Statement

"Management of Commercially
Generated Radioactive Waste"
DOE/EIS-0046-D

General Comments

Summary

The U.S. Environmental Protection Agency (EPA) has reviewed the U.S. Department of Energy (DOE) Draft Environmental Impact Statement (DEIS) for "Management of Commercially Generated Radioactive Waste" (DOE/EIS-0046-D). This DEIS replaces one (WASH-1539) that was issued in September 1974, by the Atomic Energy Commission (AEC) and withdrawn in April 1975, by the Energy Research and Development Administration (ERDA). It is clear that the comments of EPA regarding WASH-1539 were seriously considered by those who prepared this Draft EIS. After correction of the errors we believe the Final EIS will support a program leading to the safe, long-term disposal of radioactive waste.

We agree with DOE that the option selected for implementation appears to be the best of those considered; however, we believe that more information should be presented on the other nine options. We believe that the DEIS has many errors; nevertheless, we doubt that the correction of these errors will show that any other option is preferable to the mined, geologic repository. It is also unlikely that there would be any viable alternative available in the near future. For this reason we believe DOE's program should be vigorously pursued.

1. Other Options Need Fuller Treatment.

The document Draft Environmental Impact Statement on "Management of Commercially Generated Radioactive Waste" considers ten alternative methods for disposal of high-level nuclear waste. However, all of the alternatives considered, except the mined, geologic repository alternative, are presented only in a general fashion. It is impossible to compare the options adequately, because detailed information is presented on only one of the options. Regulations developed by the Council on Environmental Quality specifically require agencies to "Rigorously explore and objectively evaluate all reasonable alternatives, and for alternatives which were eliminated from detailed study, briefly discuss the reasons for their having been eliminated" (40 CFR 1502.14 (a)). The difficulty is clearly displayed by the nature of Section 4.0, "Comparative Analysis of CWM Options." The DEIS appears inadequate to support decisions which would eliminate alternatives from the scope of a permanent disposal strategy. For example, Table 4.5.1 suggests that the "Very Deep Rock" concept is preferable to "Geologic with Resynthesis" for virtually all decision criteria. This would suggest dropping the resynthesis alternative. However, we do not feel that there has been sufficient analysis to justify such a conclusion. The final EIS should improve the comparative analysis as much as possible and should specify to what extent the program will pursue the nine alternatives other than the mined, geologic repository option, which has been selected for implementation. DOE should also consider changing the title of the Final EIS to "Management of Commercially Generated High-Level and TRU Radioactive Waste," in order to better reflect the thrust of the document.

A "no action" alternative should be presented. Although this alternative is neither socially nor environmentally acceptable, it would be useful to present this option. To some extent this alternative is discussed as the delayed decision options in Section 3.1. This specific alternative is required by 40 CFR 1502.14(d).

We believe the "Chemical Resynthesis" approach deserves further detailed consideration. The possibility of using a waste form which is thermodynamically stable, does not form metamicts over time, and is almost entirely insoluble in a wide range of geological liquids offers advantages over other waste forms, because the population dose from all events except intrusion and catastrophic releases (volcanism, meteorite breach, etc.) is very low. Several references in this field have been

published since the Draft EIS was prepared. (for example, those of Ringwood, McCarthy, and Roy). As noted in the EIS, this option is really a variation of the geologic disposal option.

2. Sensitivity Analysis is Needed.

In describing the impact from the mined geologic repository, the DEIS uses probability and consequence parameters which are currently very uncertain. The impacts calculated in the DEIS are frequently more pessimistic than results we have obtained in our program to develop environmental standards for HLW. Different assumptions about many parameters may have significant impact on the projected health risk. Consequently, we strongly recommend that the Final EIS contain sensitivity analyses to indicate the range of impacts which may result from varying these parameters. The uncertainties in these analyses should also be identified and discussed.

3. Dose Calculations Need Improvement.

There are substantial problems in the calculation of radiation doses and health effects to the public. The time-integrated population dose is frequently neglected. Furthermore, population doses are not always expressed as fatal, non-fatal, and genetic health effects; we think that they should be. The DEIS suffers technically from old references, occasional misquoting of data, and some lack of balance in presenting radiobiological concepts. The use of old estimates of natural background in developing risk perspective and the use of dated and/or questionable references coupled with a lack of balance in presenting radiation risk coefficients result in a less than satisfactory draft for public decision making. Additionally, the methodology used for the impact assessment is in need of improvement. (See specific comments for Appendices C, D, E, H, and I.)

The Draft EIS appears to indicate that the major hazards occur in the first few hundred years while Sr-90 and Cs-137 are present. As a result, long-lived nuclides, such as Tc-99 and I-129, are neglected despite the fact that they can be geochemically mobile under some circumstances. One of the major shortcomings is the fact that only one set of sorption constants has been used in this work. (This assertion is supported by a caveat on page 3.1.160.) Sensitivity analyses performed in our high-level waste program indicate that changes in sorption constants and other parameters lead to significant changes in time-integrated population dose and dose to maximum individual. Recent work suggests that the impact of some nuclides is controlled more by solubility considerations than by sorption considerations (See, for

example: Bondietti and Francis, Science, 203, pp. 1337-1340, 1979). A range of sorption constants and solubility limitations for specific elements should be used.

4. EPA will soon publish Criteria and Standards.

Our approach to the problems of radioactive waste disposal is to use as many independent barriers as feasible to reduce the environmental impact, taking into account social, technical, and economic factors. We are presently preparing criteria and standards for radioactive waste management. The criteria (applicable to all radioactive waste) were published in draft form last year (43 F.R. 53262 et seq.). The comments received from other agencies and the public are presently being analyzed for final publication in several months. The high-level waste standard now under development will also be published in draft form in several months. These criteria and standards will be applicable to any disposal of high-level waste or spent nuclear fuel.

Economic Comments

1. The Effects of Costs Are Not Considered.

A major fault in the DEIS is that the potential economic impact resulting from the cost of commercial waste management is not addressed. Since payment of these costs will be made by the consumers of nuclear-generated electricity, it is necessary to determine what the impact will be on electricity customers. The DEIS estimates the cost of waste management but does not evaluate the economic consequences of incurring such costs. Analysis of both the microeconomic and macroeconomic impacts should be performed. Within the micro framework, the direct impacts on customers' electric rates and fuel bills should be investigated. Macroeconomic considerations should include the degree of secondary impacts stemming from a rate increase to commercial and industrial electric users which can influence the cost of producing other goods and services in the economy. The economic impacts of the cost of waste management also need to be discussed on a regional basis since they depend on each area's relative reliance on nuclear-generated electricity. The potential for these costs to influence the selection of power plant type should also be addressed. In light of the relatively detailed analysis of the localized socioeconomic impacts associated with the siting of waste management facilities which was presented in the DEIS, the lack of an economic impact analysis of the cost of waste management is a serious omission in the report.

2. Impacts on Nuclear Power Growth Are Not Addressed.

Another fault of the DEIS is the stated intent to exhibit neutrality regarding nuclear growth in connection with analysis of the effect of deferring the repository startup date to the year 2000. By assuming that there is no relationship between deferral of the repository startup and nuclear power growth, the analysis generates misleading results about the impacts of the deferral. By recognizing the existence of administrative and legislative obstacles to nuclear expansion, which are tied to the absence of a demonstrated waste management plan (e.g., the California moratorium), one must conclude that the deferral of a repository startup date for 15 years should result in a lower forecast of nuclear activity. As the DEIS indicates, different levels of nuclear activity produce different degrees of environmental and economic impacts. There is no objection, per se, to determining the respective impacts of two different situations which use the same nuclear power forecast but different repository startup

dates. The objection arises when the inference is made that the effect of the deferral is simply the differences in the estimated parameters for each situation. The differences in these two cases should not be interpreted as representing the impact of deferring the repository startup date, since deferral necessitates a different (lower) level of nuclear activity with its accompanying level of environmental and economic values. Thus, the true impact of deferral must be estimated by varying the nuclear power forecast from the base case (1985 startup date).

The DEIS misleads the reader since the impact of deferral is presumed to be estimated from Table 3.1.84 which summarizes the environmental effects for alternative repository startup dates of 1985 and 2000 (see page 4.42, second paragraph). On page 4.45, second paragraph, it is stated that the variations in health and safety effects as well as cost impacts by different strategies, which include deferral of repository startup date, are small. Despite the caveat stated in the foreword about neutrality regarding nuclear growth, by utilizing this neutrality in the estimation of the environmental effects the DEIS has incorrectly estimated the impact of deferral. A proper estimation procedure must address the fact that the forecast of nuclear power growth is dependent (among other things) on the repository startup date.

Specific Comments

1. (Page 1.5, last paragraph, line 8) The bentonites referred to are sodium montmorillonites which lose water when heated to 100°C. Although the adsorption of metals is high in bentonites, the water release is an undesirable property in proximity to canisters. Perhaps illite could be utilized in lieu of bentonite to the assured 100°C isotherm.

2. (Page 1.9) Why is the statement made that "in either event, the HLW contains fission products, uranium, plutonium, and the balance of the TRUs?" In both the recycle options most of the uranium is removed, and in the U-Pu recycle most of the plutonium is removed as well. Furthermore, if one assumes U & Pu recycling, sooner or later one reaches the point where fuel elements no longer have sufficient fuel value to be worth recycling. This case should be considered.

The volatile materials and TRU elements separated in fuel processing and captured in accordance with the uranium fuel cycle standards (40 CFR 190) are omitted in this discussion. They should be included.

3. (Page 1.11) In comparing natural and manmade doses person-rem is the sum of doses to individuals in the population and so is a function of both individual doses and population size. The extra 260,000 person-rem in Colorado compared to Louisiana is meaningless in that population size selection was arbitrary. Why not use New York and Hawaii? Moreover, the data base is now obsolescent, see NCRP-45.

4. (Page 1.12) Media properties: This section appears to consider only the properties of a medium which make it possible to construct a mine in it. For example, ground water is discussed only in terms of the necessity to remove water from repository shafts and rooms. Ground water is more important as a potential way for the radionuclides to move in the geosphere.

5. (Page 1.15) Again bentonite is considered despite the limitations expressed above.

Under human institutions: Human back-up of the "carefully engineered geologic system" for periods of hundreds of years is not enough. Back-up for thousands of years, if not many thousands, would be necessary. It is for this reason that the proposed EPA criteria (43 F.R. 53262 et seq., November 15, 1978) recommend against reliance on institutional controls for more than one hundred years.

The hazard indices discussed in Section 3.1.3.4 and mentioned here are at best crude estimates. The hazard of a material is based on three factors--(a) the quantity of the materials available, (b) the toxicity of the materials, and (c) the pathways between the materials and human beings. Hazard indices which do not consider all of these (and it is difficult to think of a generic hazard index which would be useful for specific pathways) are not particularly useful.

6. (Page 1.19)--Accident analysis: The impacts associated with accidents after closure of the repository have been improperly assessed. The most suitable assessment measure is the time-integrated population dose over the time of interest which, for many accidents involving ground water flow, would be a very long time. Maximum individual doses would probably be associated with the ingestion and use of ground water; this was not calculated.

The chemical nature of the waste and of the geosphere appears to be largely ignored. Much of the reduction of radiation dose appears to occur as a result of the delay of radionuclides by sorption from the ground water. The sorption of radionuclides depends on several factors, including the oxidation-reduction state of the nuclide, the presence or absence of complexing or chelating agents, and the nature of the specific geological materials present. In some cases, particularly if large and rather exotic containment canisters are postulated, the ion exchange requirements of the canister materials may be quite significant and might overload the exchange capacity of the media in which the waste was emplaced.

7. (Page 2.2.1, Section 2.2.1.1) The document neglects to mention overall guidance provided by the FRC: Radiation Protection Guidance for Federal Agencies, 25 F.R. 4402 et seq. (5/18/60), for which 10 CFR 20 is one of several implementing regulations.

8. (Section 2.2.1.2, page 2.2.3 et seq.) There is no mention of EPA's regulations developed under the regulatory authority of the Marine Protection, Research and Sanctuaries Act of 1972 (Public Law 92-532). This authority should be referenced in this section.

9. (Page 2.2.3) Under EPA Uranium Fuel Cycle Standards, the last sentence in this section is in error. The effective date for application of 40 CFR 190 can be found in 40 CFR 190.12. This error should be corrected.

10. (Page 2.2.3) The way in which the EPA drinking water regulations would be applied, if at all, is not made clear. These regulations are not directly appropriate to the disposal of radioactive waste since they do not control the contamination of the environment. They are

directed toward a water supplier and applied to monitoring and corrective treatment regardless of the source of the contamination. The Draft Environmental Impact Statement relates to activities of persons whose contamination of the environment is being limited.

11. (Page 2.2.4, Line 6): This should be corrected to read: (b) Gross alpha particle activity (including radium-226 but excluding radon and uranium)--15 pCi/l.

12. (Page 2.2.5) Under "Clean Air Act Amendments of 1977" the text states: "The administrative and legal problems arising from the potential conflict with NRC regulatory authority and procedures originating in the Atomic Energy Act of 1954 have not been resolved. However, it is unlikely that existing EPA radiation standards will be changed, although administrative requirements may." This statement is presumptuous and does not reflect the major effort underway at EPA to develop regulations under the Clean Air Act, as amended. The text should be revised in the Final EIS.

13. (Pages 2.3.2 and 2.3.3, Tables 2.3.1 and 2.3.2) Both of these tables are taken from an obsolescent reference (ORP/CSD 72-1). More appropriate references would be EPA report ORP/SID 72-1 (reference 21, Section 2.3) with the cosmic ray doses augmented by the new information in NCRP Report #45 (reference 10, Section 2.3).

14. (Page 2.3.4, Section 2.3.2.2, 2nd paragraph, last three lines) The dose estimates for radon are obsolete. Currently, dissolved radon in the body would give a dose of about 2 to 3 mrem/yr and the range of estimated dose from inhaled radon and daughters at 0.7 pCi/liter would be 130 mrem/yr to 1800 mrem/yr. See United Nations Scientific Committee on the Effects of Atomic Radiation Report (UNSCEAR) for 1977 for a more complete treatment of the question, also NCRP #45.

15. (Page 2.3.5, Section 2.3.2.2, Table 2.3.3)-The data in this table is obsolete, see UNSCEAR 1977 or NCRP 45 for current data.

16. (Page 2.3.5, Section 2.3.2.2, Table 2.3.4) - This table is obsolete. NCRP 45 summarizes natural background as follows:

Annual Exposure in Millirem/year

Source	Bone				GI Tract
	Gonads	Lung	Surface	Marrow	
Cosmic	28	28	28	28	28
Cosmogenic	0.7	0.7	0.8	0.7	0.7
Terrestrial	26	26	26	26	26
Inhaled Radionuclides in Body	--	100	--	--	--
TOTAL	80	180	120	80	80

17. (Page 2.3.6, Section 2.3.3, 2nd paragraph, last line) The use of the Congressional Research Service (reference 23) estimate of 200,000 defective children per year does not agree with current estimates of 9.5 percent to 10.5 percent incidence of genetic disorders in newborn (see UNSCEAR 1977, p. 519). The UNSCEAR estimates suggest that this estimate of 200,000 is at least a factor of 2 low.

Estimates of malignancies occurring each year are better obtained from the American Cancer Society annual publication "Cancer Facts and Figures--19xx." For example, estimates have been: 395,000 deaths, 765,000 cases of cancer--1979; 390,000 deaths, 700,000 cases--1978; 385,000 deaths, 690,000 cases--1977; etc.

18. (Page 2.3.6, Section 2.3.3, 3rd paragraph, last sentence) The use of Frigerio and Stowe as a reference should be put in context. Aside from using the same obsolete reference of natural background used in this DEIS which inflates the probable difference in background between areas of the country, the authors neglect to consider the potential effects of other carcinogens in the work place and the environment. Some of these problems are highlighted in multi-author sections on "Demographic Leads to High-Risk Groups" and "Environmental Factors" in the volume Persons at High-Risk of Cancer (J.F. Franmeni, Jr., editor, Academic Press, New York, 1975). Little support is given for the assertions in the referenced paper.

In a more complete report by the same author (N.A. Frigerio, K.F. Eckerman, and R.S. Stowe, "Carcinogenic Hazard from Low-level, Low-rate Radiation," ANL/ES-26, 1973) where all methods and assumptions are given, there are several flaws. A major flaw is the assumption that "all forms of cancer show very similar doubling doses and closely similar increases in mortality rate per rad." This assumption is made

contrary to the evidence in ICRP, UNSCEAR, BEIR, and other reports that variations in the susceptibility of tissue to induction of different forms of cancer by irradiation are quite large and not necessarily related to the marked variations in natural incidence of the diverse types of cancer.

There are also problems in the statistical analysis in ANL/ES-26: misuse or misinterpretation of the t-statistic, failure to use Scheffes' test or calculations of variance ratio to check the significance of the series of t-tests, and use of gross averages in the analysis.

In reality, the paper can be shown to be erroneous by inspection of Frigerio *et al.*'s source of cancer mortality data, NCI Monograph 33, Patterns in Cancer Mortality in the United States: 1950-1967. In Monograph 33, Burbank presented an analysis of Dynamic Geographic Distribution for each cancer. The complex pattern of increasing and decreasing cancer mortality by state and cancer show that factors other than background are the major driving force in cancer mortality rates and that natural background radiation is not.

Indeed, in a later publication (Jacobson, A.P., Plato, P.A., and Frigerio, N.A. "The Role of Natural Radiations in Human Leukemogenesis," Am. J. Public Health, 66, p. 31-37, 1976), a more reasonable major conclusion was reached: "It appears that conditions relative to populations and their environment could mask a radiation effect, if in fact one is present."

19. (Page 3.1) The main reason that the waste management system evaluation was emphasized on deep geological disposal in salt formations was that this alternative received the most study. We suggest other geologic media and other disposal methods receive adequate scrutiny.

20. (Page 3.1.3) The concept of the equilibrium release fringe does not appear to have significance for long-term isolation, considering the potential for disruption of the containment.

21. (Page 3.1.4, last paragraph) We recommend that the last paragraph on this page (mid-paragraph) address the fact that most of the intense tectonic activity and virtually all of the volcanic activity of the North American continent occur along the global plate boundaries.

22. (Page 3.1.6, sixth paragraph) This discussion of resource potential of the host rock is incomplete. It treats only the loss of resource through use of the area for a repository. In fact, such loss of resource potential cannot be maintained indefinitely since eventual loss of control over this site must be expected. The real problem with resource potential of the host rock is that it will tend to attract future human intrusion and so increase the probability that the repository integrity can be breached by man's activities.

23. (Page 3.1.8, last paragraph) We recommend that the low water content and drier nature of salt domes resulting from the diapirism process be cited as a particular advantage over bedded salt.

24. (Page 3.1.14 - Table at bottom of page) There should be two categories of salt, Bedded Salt and Salt Domes, so that the difference in moisture content can be emphasized. Salt Domes have lower moisture content which is a major consideration.

Plasticity, ion exchange capacity, and linear discontinuities should be added to the properties for the rock types. Plasticity values would have the following relative scale values: Bedded Salt (3), Salt Domes (3), Granite (1), Shale (2), and Basalt (1). Ion exchange capacity would have values of 1, 1, 2, 3, and 2, respectively. Linear discontinuities would have values of 3, 3, 2, 3, and 1, respectively.

25. (Page 3.1.16, third paragraph) This statement is too vague to be useful in the site selection process. It is necessary to make a determination as to the period of time for which high-level radioactive waste must be kept isolated from the biosphere and also for how complete such isolation must be. Complete isolation for all time is probably not achievable and certainly cannot be assured in advance.

26. (Page 3.1.17) Somewhere it should be emphasized that global plate boundaries should be excluded as locations for potential repository sites because most of the catastrophic geologic events of this planet occur along these well defined linear features.

27. (Page 3.1.17) Several uncertainties exist in the projected behavior of the system, such as the philosophy for radionuclide containment, waste form, the host rock, etc. We suggest there also be a discussion as to how these uncertainties can be overcome.

28. (Page 3.1.20) How important is climatic change in determining the evaluation of the hydrologic environment? Increased precipitation might fill up an unsaturated aquifer, but how would it affect the permeability or porosity of an aquifer in the zone of saturation?

29. (Page 3.1.22, first paragraph) The first sentence should substitute volcanic activity for magmatism and a sentence immediately following should cite the fact that unlike other catastrophic events 98 percent of all volcanism on earth is confined to regions of global plate boundaries. Since this is a known fact, the reference to magmatism in line 11 should be deleted. A sentence following line 11 (with magmatism deleted) could state that volcanism (magmatism) is predictable since it is virtually confined to the region of global plate boundaries. In any case volcanism (magmatism) should be clearly separated from tectonism and this has not been accomplished on page 3.1.22.

30. (Page 3.1.22) Isotopic date provinces have been delineated in the Precambrian basement rock (Dott and Batten, Evolution of the Earth, 1976, p. 165), and the importance of these boundaries may be second only in significance to the global plate boundaries in site selection. While these boundaries are exposed in the Canadian Shield they are buried beneath sedimentary rocks in the rest of the North American continent; however, the extension of the Grenville isotopic boundary beneath the sediment has been correlated with major seismic events in the eastern portion of the North American continent. It is recommended that the isotopic date boundaries be addressed in site selection considerations.

31. (Page 3.1.23, third paragraph) It should be recognized that both climate and hydrologic gradients may change in the course of time.

32. (Page 3.1.24, middle) There is a reference to a nonexistent section 3.1.1.3.

33. (Page 3.1.24, seventh paragraph) It is not necessarily true that the primary geological barrier to waste migration will be the repository host rock. Intrusions by man or overriding natural processes and events may throw the primary dependence onto other geological formations.

34. (Page 3.1.25, bottom half) This material is of concern only in the short term.

35. (Page 3.1.26, third paragraph) This appears to be largely a list of problems, with only speculative solutions. How does DOE intend to cope with these problems?

36. (Page 3.1.27, line 19) The spelling of "intrusive" rocks should be corrected to "intrusive."

37. (Page 3.1.28, first paragraph) The measurement of the equivalence of the hazards of waste repositories and natural ore is a complex subject. Many of the hazard indices are concerned only with the amount of radioactive material and its toxicity, without consideration of routes by which the activity can reach man.

38. (Page 3.1.32) The discussion of ground water flow appears to be concerned only with maintaining a dry repository during the construction period. It is necessary to consider the ground water after the closure as a possible means of transport of the radioactivity.

39. (Page 3.1.35, fourth paragraph) There is an implication in this paragraph that the unknown problems listed will have an effect only on the cost of the repository, requiring generally greater spacing and therefore a larger, more expensive repository. The question of reduction of the effectiveness of the repository, for example by fracturing rock, is not addressed.

40. (Page 3.1.38, first paragraph) This paragraph suggests that the fission product problem terminates with the decay of strontium-90 and cesium-137. While this may be substantially true for the production of heat and for acute radiation hazard, it is not true for the significance of the waste as a health hazard. Doses from technetium-99, iodine-129, and cesium-135 are not negligible over a long time frame. This problem is repeated in the last sentence of the next paragraph.

41. (Page 3.1.38, fifth and sixth paragraphs) In the early phases, actinide elements, particularly plutonium-238 and plutonium-241, are significant. Tritium may also be significant. In the long time frame cesium-135 and carbon-14 might also be significant.

42. (Page 3.1.40, licensing) Does the waste package design refer to the container alone, the container plus waste form, or the entire system? It does not seem that the waste container and form problems are more difficult from a licensing standpoint than the other barriers.

43. (Page 3.1.40, physical protection) This part makes sense if it refers to physical protection during the operational phase of the repository. The first paragraph on page 3.1.41 is self-contradictory. It states that the waste would be essentially unavailable after placement in the geological repository. Because the operational controls will cease to exist long before any appreciable decay of plutonium-239, the protection must be inherent in the inaccessibility of the waste in the repository and the massive effort that would be required to remove it.

44. (Page 3.1.41, third bullet) Comparison of acceptable consequences from radiation in comparison with those from automobile accidents is invalid. There are two primary aspects to the establishment of bases:

1. How much will society accept on an absolute basis;
2. How much better than this can the technology provide?

45. (Page 3.1.41, last bullet) The use of adsorption coefficients from one set of Hanford subsoils, measured under laboratory conditions, is not an adequate basis for scoping the effect of adsorption. There are some substantial differences between the adsorption coefficients of the Hanford Subsoil and of those given on page K-20 of the Waste Isolation Pilot Plant EIS (DOE/EIS-0026-D), for example.

46. (Page 3.1.47 - 5th paragraph, last sentence) Isotopic age province boundaries should be added to the list of areas to be avoided in the preliminary selection of repository areas.

47. (Page 3.1.48, near bottom) The first statement under ground water implies that a repository can be sited in conjunction with a useful ground water source without affecting its quality. Since EPA analysis has shown that ground water can be significantly contaminated by a single drilling event, it is important that the repository should not be situated where it can affect a useful ground water resource.

48. (Page 3.1.49) The chemical nature of any aquifers around a repository should be briefly discussed. Reducing aquifers greatly limit the solubility of U, Pu, Np, and Tc. Sorption of these elements would provide a further reduction in the amount of nuclides reaching people, and sorption of the reduced states of these elements is higher than sorption of the oxidized states. This oxidation-reduction consideration should be briefly discussed.

49. (Page 3.1.49, third paragraph) "Flow rates and velocities of ground water that are insignificant over a 50 year period will have to be considered in terms of hundreds to thousands of years." This statement should be further discussed.

50. (Page 3.1.52, item four) It seems unreasonable to limit the search for an optimal site to areas with "availability of title". Surely this is an area where eminent domain is appropriate.

51. (Page 3.1.52, last sentence) There is an unwarranted belief that all problems can be solved by major efforts. Investigations into a basic research area, such as this appears to be, do not necessarily have satisfactory outcomes. The research must first be performed before opinions as to its results are valid.

52. (Page 3.1.54, item two) The chemically separated high-level waste to be considered must include the iodine-129 (and the other volatiles and transuranics) which are excluded from discharge to the atmosphere from the fuel reprocessing plant by EPA's Uranium Fuel Cycle regulations (40 CFR 190.10(b)).

53. (Page 3.1.54, item three, last sentence) Unless the actinides from chemically separated high-level waste are recycled, they must be disposed of as waste and would still require consideration in this Environmental Impact Statement. Even if one assumes recycling of uranium and plutonium, one eventually reaches the point where recycling is not economically feasible and the transuranics must then be disposed as waste.

54. (Page 3.1.55, second paragraph) The leach rates of spent fuel in room temperature deionized water are irrelevant. The leach rates of spent fuel in typical ground waters at temperatures to be expected in spent fuel repositories are more important.

55. (Page 3.1.59, second paragraph) The canister could prevent ground water intrusion for a period provided that there was no disruptive event which would destroy the canister. Such disruption would be expected in gas, oil, or mineral exploration. It therefore seems that it would be impossible to maintain canister integrity from credible accidents for a significant time period.

56. (Page 3.1.59, third paragraph) Reliance on the canister alone for long-term containment seems unwarranted, as indicated in the previous comment. This does not mean that canisters which would be durable in the absence of an intrusive event or traumatic geological events should not be developed.

57. (Page 3.1.62, first paragraph) The likelihood that oxygen will be introduced into the repository when it is constructed and therefore be available to the ground water should be considered in evaluating canisters. The mobility of some nuclides is also increased by the presence of oxygen. The possibilities that the copper would be a resource which would attract human intrusion should also be considered.

58. (Page 3.1.62, fourth paragraph) The misspelling of alumina should be corrected.

59. (Page 3.1.62, Section 3.1.3.3) This section appears to assume long-term institutional functioning. Expectation that institutions will continue over thousands of years (or even that they will maintain their control over hundreds of years) is not well-founded.

60. (Page 3.1.64, first paragraph) The first century after closure of the repository would be critical for "hands on" corrective action only if the monitoring program established some deficiency in the repository. Although the radioactivity of the repository has been reduced substantially after 700 years, the threat is by no means negligible.

61. (Page 3.1.64, second bullet) What evidence is there that technical information can be maintained for a very long time? What constitutes "a very long time"?

62. (Page 3.1.64, fourth paragraph) The definition of risk as "the sum product of the magnitude of losses and the probability that the losses will occur" is questionable. There is a tendency for aversion of high consequence accidents, which would imply a valuation other than a strict p-c product.

63. (Page 3.1.64, seventh paragraph) This whole paragraph is very questionable. Hazard indices are not based on estimates of societal risks compared to other societal risks, in general. There is also the question of whether the hazard of the waste after several hundred years of decay, considering nuclides and pathways, is less than the hazard of the ores.

64. (Page 3.1.64, last paragraph) Consequence analysis for any release is the estimation of the effect of that release. It is not restricted to postulated worst cases.

65. (Page 3.1.65, fourth paragraph) The total quantity of radioactivity in curies is irrelevant. The nature of the radionuclides and their pathways to man are significant.

66. (Page 3.1.67, first paragraph) Are these probabilities best estimate probabilities, upper bound probabilities, or what?

67. (Page 3.1.67, second paragraph) Is ^{229}Ra a misprint for ^{226}Ra ?

68. (Page 3.1.67, third paragraph) Since the long-term behavior of the parameters is uncertain, risk assessment should be based on upper estimate predictions as well as on "reasonable" predictions.

69. (Page 3.1.71, seventh paragraph) Sorption of radionuclides is controlled by the site-specific geology. It seems unlikely that radionuclide behavior data from one site can be applied to another site.

70. (Page 3.1.100, first paragraph) The destruction caused by a meteorite striking one of our large metropolitan areas is irrelevant to this consideration. We have no control over where a meteorite will fall; therefore, one place is as good as another, and the possibility of a meteorite strike does not become a consideration in the location of cities. The probability that a meteorite will disperse materials from a deep geological repository is controllable in that the probability of a meteorite large enough to cause disruption is a function of depth and can be reduced as much as desired by going deep enough.

71. (Page 3.1.100, sixth paragraph) The regional population for the types of releases considered most likely in the long term in waste disposal is the population in a river basin rather than the population within an eighty kilometer radius of the plant.

72. (Page 3.1.101, second paragraph) Why is bone an organ of principal interest? According to the BEIR work, more health effects would be expected from a dose to red marrow than from the same dose to bone. It is also probable that the liver should be considered a significant organ.

73. (Page 3.1.105, first paragraph) The Arthur D. Little work for EPA found that spent fuel heat loadings should be about the same for granite and salt. This seems reasonable considering that the salt has a higher conductivity than the hard rocks and is surrounded by shale which is not a good conductor. We would like to correct this discrepancy between the DOE and A.D. Little heat loading models.

74. (Page 3.1.106, third paragraph) There is no discussion of the air content of the repository following backfilling. The mobility of several significant radioelements, plutonium, neptunium, uranium, and technetium, are affected significantly by their oxidation state. (In some schemes for the in situ solution mining of uranium, air is used as the source of O₂ and is the oxidizing agent of the uranium.) The air content should be briefly discussed.

75. (Page 3.1.111, last paragraph) There should be some consideration of the possible interaction of the various wastes with each other. If the transuranic waste contains organic material, these may contain chelating materials, which could have an effect of mobilizing other waste.

76. (Page 3.1.129) Social service demands (Table 3.1.21) were derived by applying factors "to the project in-migration values" (Table 3.1.19). Therefore, the level of forecasted social service demands by individual site should be proportional to the estimated level of project in-migrants for each site. From Table 3.1.19, under the maximum impact condition the respective estimates for the number of project in-migrants for 1985 indicate the lowest value for the Midwest site (5800), followed by the Southeast site (8600) and the Southwest site (15,000). However, in Table 3.1.21, also under the maximum impact condition, some of the social services--physicians and dentists, and hospital and nursing care beds--indicate values for 1985 which reverse the relative position of the Midwest and Southeast sites. This apparent error occurs in similar tables throughout the DEIS.

77. (Page 3.1.129) The following statement appears to be incorrect in light of the information presented in the accompanying tables:

"Although the numbers of in-migrants are smaller, the potential for impacts in the Southeast maximum impact condition is quite similar to the potential in the Southwest site under maximum conditions. This is the case because the base population in the Southeast is roughly twice that in the Southwest site; therefore the Southeast is capable of absorbing greater population influx, other things being equal."

It appears that the words "Southeast" and "Southwest" should be reversed, since if the number of in-migrants for site A is half the number of site B, and the number of in-migrants stated as a percent of each site's base population is the same for each site, then the base population of site B must be twice the base population of site A. Site B would be more capable of absorbing population influx.

The identical statement is repeated in other portions of the DEIS (p. 3.1.180 and p. 3.1.194) in referring to other estimates of the numbers of in-migrants associated with different types of waste management facilities.

78. (Page 3.1.137, last paragraph) The ground water releases did not appear to include the "two aquifer case," which is most significant for ground water releases. This case involves a hydrologic connection between two aquifers through a repository, with subsequent ground water transport. The analysis also appears to consider only release of the total radionuclide content, which does not appear to be a credible or useful form of analysis.

79. (Page 3.1.138, second paragraph) The presence of salt would probably not preclude the use of the water as a source of food or recreation. The salt would be diluted to acceptable levels by any reasonable amount of water far more quickly than the radioactivity.

80. (Page 3.1.139, third paragraph) What are the bases for assuming that 10 percent of the particulates suspended are of respirable size?

81. (Page 3.1.140) How is the uranium-238 depleted by so much over a period of one thousand years?

82. (Page 3.1.147, fourth paragraph) A meteorite of the described size would undoubtedly produce a local disaster area. The impact of the meteorite, however, would also disperse radioactive materials into the atmosphere from which they would impact over an extended area. It is the additional impact of this radioactive material that is significant. It is not likely that the impact would be local or that it could be controlled by local monitoring.

83. (Page 3.1.148, fourth paragraph) This scenario does not appear to be a particularly bad case because of the limitation of the contact for one year. Such a limitation, together with slow leaching, results in a minimal release of radionuclides. What is the effect of continued erosion?

84. (Page 3.1.149, table) A leach rate of 10⁻⁴ g/cm²/day, applied to a one centimeter cube of density 2, would result in a leaching rate of 3x10⁻⁴ per day, or approximately 0.1 per year. Is this the value that was used in the analysis?

85. (Page 3.1.150, fifth paragraph) Doses to a maximum individual are not the best measure of the impact of this accident. Total population

dose and integrated population dose are more significant. It might be noted that the emergency dose limits of 100 rem and 25 rem apply implicitly to the case where only one or a few people are exposed.

86. (Page 3.1.155, last paragraph) This is an improper combination of probabilities. If the probabilities are multiplied together, as has been done here, the result is the probability of all three conditions occurring in the same year. If the probabilities are taken over 10,000 years, for example, the probability of a fault intersecting the repository is 4×10^{-7} . The probabilities of failure of waste containment, or of aquifer intersection, over this period are likely to be one, each. The total probability is therefore about 4×10^{-7} , not 4×10^{-15} .

87. (Page 3.1.157, table) Dose commitments of 10^8 person-rem are estimated to result in 2×10^4 fatal cancers. The risk associated (including the probability) is much less.

88. (Page 3.1.161, second paragraph) The conclusion drawn from the comparison with the ore body would be improved if some analysis were provided.

89. (Page 3.1.162, last paragraph) Distribution of the waste would lower the maximum and regional individual doses, but would increase the probability of the event by a factor equal to the number of repositories.

90. (Page 3.1.163, second paragraph) It is not proper to assume doses to the regional population from a ratio basis with total body dose.

91. (Page 3.1.164, second paragraph) Diverting the entering stream until it is diluted by another stream does not change the population dose. It merely means smaller doses to more people.

92. (Page 3.1.165, third paragraph) The overall probability of a contaminated drilling event can exceed 0.005, since there is a chance of more than one drilling event over a period of time.

93. (Page 3.1.191, table) The quantity of krypton-85 released to the air should be related to gigawatts of electricity produced for comparison with uranium fuel cycle standards.

94. (Page 3.1.119) There is an inconsistency in the unit cost of spent fuel storage stated in Table 3.1.82 and the text (2nd paragraph)

on p. 3.1.119. In Table 3.1.82, the cost for the cooling of spent fuel is based on 75 percent of the spent fuel being stored in reactor basins at \$30/kg. On p. 3.1.229, it is reported that 75 percent of spent fuel storage requirements are provided by power plant basins at an average cost of \$6/kg HM-yr, which would amount to \$36/kg for six years. Recalculating the spent fuel storage cost based on the assumption stated on p. 3.1.229, the cost of six-year cooling of spent fuel in Table 3.1.82 should therefore be \$44/kg instead of \$39/kg.

95. (Page 3.1.229, bottom) There is a brief discussion on the effects of using present worth costs as opposed to undiscounted costs. It should be mentioned that the relative ranking of the fuel cycle alternatives by their total costs is significantly different according to which set of costs is used. As indicated by Tables 3.1.89 and 3.1.90, the U & Pu Recycle under the Deferred Fuel Cycle Decision has the highest total cost, about 50 percent higher than the least expensive case, on the basis of undiscounted costs, while this same case possesses the lowest total cost of any alternative when discounted (at a 7 percent rate) costs are used.

96. (Page 3.1.242, third paragraph) The study of rock-waste interactions should include the geochemistry. Mobility of a number of radionuclides is strongly affected by the geochemistry (particularly the oxidation-reduction potential of the repository and ground water) and by the potential presence of complexing agents. These should be included in the proposed research program.

97. (Page 3.2, Section 3.2) This entire section appears to be largely speculative. The comment as to the speculative nature of the discussion applies to Sections 3.2, 3.3, 3.4, and, in fact, all the rest of Chapter 3.

98. (Page 3.2.2, 3rd paragraph, line 11) The word pegmatite should be replaced by migmatite. The reference (Leonardos, 1974) specifically states that "pegmatite contribution to monazite deposits are trivial" (p. 1126). On 1127, Leonardos states "migmatites have supplied the material for the bands within the quartzite". Thus, migmatite is the term required to support the reference.

99. (Page 3.2.3, second paragraph) This paragraph does not consider the effect of radiation damage in the glass. This should be mentioned.

100. (Page 3.3.13, seventh paragraph) The possibility that oxygen introduced with the waste will change the reducing conditions should be considered.

101. (Page 3.6.1) The first paragraph of this section should state that at present it is illegal to put high-level wastes in, on, or under the seabed and that legislative action would be required before implementation.

The fourth paragraph states that a ship will monitor the emplaced wastes for an "appropriate length of time." How long (or short) is this "appropriate length of time?"

102. (Page 3.6.2) In mid gyre there is little benefit from deposition of sediments since this process is very slow there. The document states that less than .01 percent of the ocean floor would be used for disposal. What total area does this represent?

The last bullet under "Advantages" suggests that an advantage to seabed disposal is the lack of need to resolve Federal-State relations problems. This is not so because problems would surely arise from port use and the loading and transportation of waste to the port.

103. (Page 3.6.5, Table 3.6.1) The biological productivity of seamounts should be included in the table.

104. (Section 3.6.2.3, Page 3.6.5) There should be a mention that the philosophy behind this approach is isolation of the waste. This approach is required by EPA regulations.

105. (Page 3.6.7) Why would one expect low nuclide concentrations around a waste canister? If the canister failed, one would expect high levels.

106. (Page 3.6.7) Under "Water Column," there is a statement that bottom currents are slow and uniform. However, in Section 1.3.6 the DEIS says bottom currents are weak and variable. This inconsistency should be corrected.

107. (Page 3.6.7) Under "Basement Rocks," the fractured nature of the basalts could provide lenses for the transport of radionuclides. This should be mentioned.

108. (Page 3.6.8) Under current EPA regulations the canister must act as a barrier until the material decays to innocuous levels. The conservative calculational assumption, that the canister will release its entire inventory of wastes, does not reflect this regulatory requirement.

109. (Page 3.6.10, Section 3.6.2.8) Seabed disposal refers to a disposition of wastes and as such falls within EPA regulatory authority for the disposal of radioactive waste in, on, or beneath the ocean floor. The seabed disposal option for HLW is not legal under current domestic law. However, we think DOE should continue to study this option to see if this is an environmentally acceptable option.

Additionally, in Section 3.6.3.6, on page 3.6.22, the DEIS states, "implementation of a sub-seabed disposal program for non-HLW is now possible under EPA's ocean disposal permit program." DOE apparently believes ocean dumping and sub-seabed emplacement are intrinsically different for high-level waste and identical for low-level waste. We believe there is no legal difference between ocean dumping and sub-seabed emplacement and any difference between the two is purely semantic.

110. (Page 3.6.14) Bottom sediments in the mid-plate areas have extensive animal tracks. Furthermore, fish in these areas make extensive vertical and lateral migrations; this indicates that there is a possible pathway from the waste to people.

111. (Pages 3.6.22 and 3.6.23, Section 3.6.3.6) See comments for page 3.6.10, Section 3.6.2.8. Sub-seabed emplacement must comply with EPA regulations promulgated under authority given exclusively to EPA under Public Law 92-532, the Marine Protection, Research and Sanctuaries Act of 1972.

The fourth paragraph of this section perpetuates the notion that sub-seabed emplacement is not ocean dumping. We consider the difference between the two to be semantic.

Should high-level waste be released, it most certainly will affect other nations, contrary to the suggestions in the fifth paragraph.

112. (Page 3.6.24, Section 3.6.4.5) Port accidents occurred in the 60's during the loading of 55 gallon drums. This issue should be presented.

Is it proposed that several canisters go into one hole in the seabed, or will each penetrometer drop wherever it may, to be followed by monitoring of 9000 different holes per year?

113. (Page 3.6.25) Several ports have banned the shipment or receipt of spent fuel. Does the proposal include use of dedicated port facilities?

114. (Page 3.6.26) What is the range of error in the results of the "unverified, theoretical model" in projecting impact?

115. (Page 3.6.30, first paragraph) Is the \$15 million mentioned for R&D costs? If not, what costs does that figure represent?

116. (Page 3.6.31) Again we find the semantic difference between ocean dumping and sub-seabed emplacement. "Dumping" and "Dump" should not be in quotation marks. It is defined in the Marine Protection, Research, and Sanctuaries Act of 1972 as a disposition of materials. This misleading section should be corrected in the Final EIS.

117. (Page 3.6.32, third paragraph) The sixth option should be clarified. Considering the dollar input, what is the intent and what will be the output?

118. (Section 3.6) Somewhere in this section several other matters should be briefly considered:

How deep would the projectiles be sent? What distance beneath the sediment surface, and how far from the rock beneath? What about the concept of recovery, if unforeseen dangers are found to exist?

119. (Page 4.12) Figure 4.4.1 had been omitted. It should be included or the reference to it should be removed.

120. (Appendix C) This appendix is grossly unsatisfactory. It concentrates heavily on doses to individuals and does not appear to recognize that more recent standards, although they may be expressed in terms of dose to the maximum individual, have population dose as part of their basis. Among such regulations are:

1. Limitations on releases of effluents from power reactors (Appendix I to 10 CFR 50);
2. The uranium fuel cycle standards (40 CFR 190); and
3. The drinking water standard (40 CFR 141).

The limitations on releases of krypton-85, iodine-129, and transuranic elements, in 40 CFR 190, are explicitly based on population dose.

The general thrust of this appendix is that population dose is not a concept suitable for radiation standards. This is incorrect because the concept of ALARA usually involves balancing the cost against the reduction in population dose. It is perhaps significant that this appendix does not include any of the BEIR reports but limits itself to the 1969 BEAR reports of the National Academy of Sciences. For currency, the appendix should consider additional references, e.g., references 1, 2, 10, and 16 from Appendix E, to bring the appendix up to 1977 at least.

121. (Page C.2, 1st and 2nd paragraphs) The paragraphs ending the section on "Background" and initiating "As Low As Reasonably Achievable Application" reflect some bias and a lack of candor in describing the use of risk coefficients in radiation protection. Almost all government agencies, particularly the EPA but including the NRC and the MSHA, have used or are using risk coefficients to estimate impact of radiation exposure. The ICRP (reference 11) has gone entirely to a risk based radiation protection system, using estimates of risk in optimizing radiation protection. ICRP has stated, "These risk factors are intended to be realistic estimates of the effects of irradiation at low annual dose-equivalents (up to the Commission's recommended dose-equivalent limits)" (ICRP publication No. 28, 1978). The NCRP (reference 15) seems to stand alone in its position discounting the use of linear, nonthreshold risk coefficients in radiation protection.

122. (Page C.3, Table C.1) While the table is titled "Comparison Chart of Radiation Standards," it then lists "Standards or Criteria" and references ICRP and NCRP values or reports. ICRP and NCRP reports are recommendations or suggestions which may or may not be adapted or modified and adopted by national regulatory agencies. The references to ICRP and NCRP should be deleted from the table.

It should be noted, however, that there are ICRP reports pertinent to health effects. ICRP publication 26 (reference 11) and publication 27 ("Problems Involved in Developing an Index of Harm," 1977), both provide recommendations on "acceptable" numerical risk estimates for radiation workers.

123. (Appendix D) While the calculational models employed may be adequate, in light of the uncertainties inherent in the input data, they are not state-of-the-art, as claimed. For example, the calculation of the 5 cm gamma dose as the total body dose for air immersion could be improved by the use of an existing code which specifically yields organ doses. Again, while the DACRIN code used employs the TGLD model, it does not explicitly treat the daughter products formed after inhalation as do more complete codes.

124. (Page D.8) - The model used to estimate the population dose commitment from carbon-14 is too conservative (i.e., overestimates the impact). If dilution by the Suess effect is not considered and the total number of health effects is integrated over all time, the release of 1.4 MCi (from Table 3.1.68) would result in about 5×10^6 deaths, assuming a stable world population of 6.4×10^9 people. It might be more realistic to make a comparison to the natural production of carbon-14 and associated health risk.

125. (Page E.1, last paragraph) The bias in selection of references is obvious. While the last sentence quotes the NCRP and its dislike of linear nonthreshold risk and its use in radiation protection, to maintain balance the ICRP's use of risk factors as realistic estimates (see comment on Appendix C, page C.2) for radiation protection and their use in ICRP publications 26 and 27 should also be documented. EPA's policy statement, 41 F.R. 28409 (1976), should also be noted.

126. (Page E.3, first paragraph) In the discussion of BEIR risk estimates, emphasis is put properly on the range of uncertainty. However, it should be mentioned that the BEIR Committee did report (reference 1, p. 168), "With this limitation in mind, the Committee considers the most likely value to be approximately 3,000-4,000 cancer deaths (or a 1 percent increase in the spontaneous rate)" (emphasis added).

127. (Page E.3, second paragraph) The paragraph considers only EPA's Uranium Fuel Cycle documents and states that the risk estimates there continue to be used by EPA. In reality EPA risk estimates have continued to change as new data becomes available. In addition to papers published by staff (e.g., Ellett, Nelson, and Mills, "Allowed Health Risk for Plutonium and Americium Standards as Compared with Standards for Penetrating Radiation," pp. 587-601 in Transuranium Nuclides in the Environment, IAEA, Vienna, 1976), various EPA reports (e.g., A Computer Code for Cohort Analysis of Increased Risks of Death, EPA 520/4-78-012, 1978, or Proposed Guidance on Dose Limits for Persons Exposed to Transuranium Elements in the General Environment, EPA 520/4-77-016, 1976, etc.) show updated risk estimates and how they were derived.

128. (Pages E.3 and E.5, discussion of the Reactor Safety Study, WASH-1400.) EPA's dissatisfaction with the health effects estimates in the Reactor Safety Study (RSS) is documented in reference 53. Recent

literature has done nothing to dispel our belief that the use of a dose rate reduction factor is ill advised as is the minimal plateau duration (30 years) used in the RSS.

The UNSCEAR 1977 Report suggests (except for leukemia) a 50-year expression period unless the period has been shown to be shorter or longer for a specific cancer (reference 2, par. 12, page 363).

In particular, the two major human cancers associated with radiation, lung and breast, are considered to elevate risk for the duration of lifespan following exposure (Land, C.E. and Norman, J.E., "The Latent Periods of Radiogenic Cancers Occurring Among Japanese A-Bomb Survivors," Late Biological Effects of Ionizing Radiation, 1, IAEA, Vienna, pp. 29-47, 1978; Archer, V.E., Radford, E.P., and Axelson, O., "Radon Daughter Cancer in Man: Factors in Exposure-response Relationships," Health Physics Society Annual Meeting, June 1978).

The dose reduction factor in the RSS report appears to be derived from an analysis by Mays, et al. considering ten sets of animal data from nine studies. If an additional two studies (that happen to show a reverse effect) are included in the analysis, the dose reduction factor becomes 1.7 ± 4.5 instead of the 0.22 ± 0.20 reported by Mays, et al. As UNSCEAR 1977 points out, most of the existing animal carcinogenesis data comes from observations at doses above 50 rads and that each tumor-model system has peculiarities which prevent generalizations across multiple organ systems and cancers. See reference 53 of this appendix for comments on the dose reduction factor in the RSS.

As has been pointed out by Crump, et al. (Crump, K.S., Hoel, D.G., Langley, C.H., and Peto R., "Fundamental Carcinogenic Processes and Their Implications for Low Dose Risk Assessment," Cancer Res., 36, pp. 2973-2979, 1976): "It is likely that the error in the acceptable dose associated with a simple linear extrapolation will be much less than that associated with species to species extrapolation to man from the laboratory animal data. The BEIR Report (ref. 16) recommended linear extrapolation on pragmatic grounds. The theoretical conclusions of the present paper are that linear extrapolation to low dose levels is generally valid as a realistic yet slightly conservative procedure" (emphasis added). That carcinogenesis by an external agent acts additively with any ongoing process is accepted by Crump, et al. and by Hilberg (Hilberg, A.W., "Low-Level Ionizing Radiation: A Perspective with Suggested Control Agency Options," in 10th Annual National Conference on Radiation Control, HEW Publication (FDA) 79-8054, pp. 386-391, 1979) in his allusion: "And, conversely because man is living in an environment of chemical additives and pollutants, these may set the stage for action of a very small amount of radiation exposure."

Most of the argument on the RSS report centered on low dose rate, low LET radiation. Alpha radiation dose response curves are usually characterized as both linear and dose rate independent (BEIR, 1972; UNSCEAR, 1977) or as possibly providing underestimated effects at low doses (Martin Brown, J., "Linearity vs. Non Linearity of Dose Response for Radiation Carcinogenesis," *Health Physics*, 31, pp.231-245, 1976; Archer, V.E., Radford, E.P., and Axelson, O.; "Radon Daughter Cancer in Man: Factors in Exposure-Response Relationships," *Health Physics Society Annual Meeting*, June 1978). No reports except the RSS report consider a threshold curve a viable concept.

129. (Page E.4, Table E.1) A column in Table E.1, headed "Environmental Protection Agency," purports to be the risk estimates used by EPA. They are actually averages for various risk models used by EPA in reports and therefore are not directly comparable to the other risk estimates in the table.

The estimates of 54 leukemia deaths/10⁶ person-rem listed in the table were extracted from EPA 520/9-73-003-B (reference 4). As stated in that report (p. A-14), the risk conversion factors are average values for absolute and relative risks in the BEIR Report, 1972. Moreover, they apply only to the dosimetric models used in EPA report 520/9-73-003-B.

The EPA risk for thyroid listed in Table E.1, 15 thyroid cancer deaths/10⁶ person-rem, is referenced to EPA 520/4-76-017 (reference 6). That risk estimate cannot be found the cited reference. However, on p. 96, *ibid.*, it states "... a population age weighted value of 60 thyroid cancers per million rems to the thyroid was used." A similar risk estimate is shown in Tables 45 and 46 of EPA report 520/9-73-003-C, Environmental Analysis of the Uranium Fuel Cycle. Part II - Nuclear Power Reactors, 1973. Note that these thyroid risk estimates refer to cases, not fatalities, and so do not fit into Table E.1.

130. (Page E.7, last paragraph and page E.8, Table E.3) Newcombe's estimate of ten genetic effects based on a normal incidence rate of 0.1 percent for autosomal dominant disorders has not been supported by other studies. Current incidence estimates are about 1 percent autosomal dominant and X-linked disorders, the estimate in UNSCEAR, 1977.

The genetic effects estimates attributed to the BEIR report and EPA in Table E.3 are not comparable to those given in the 1977 UNSCEAR report since they assume a 30-year reproductive generation time. To compare the BEIR and EPA estimates with those of UNSCEAR, the BEIR and EPA estimates should be multiplied by a factor of about 0.6 to adjust for a 30-year population generation vs. the current, approximately 50-year population generation. More recent EPA estimates have been adjusted for the current population generation (EPA 520/1-75-010), to yield 200 genetic effects, close to the UNSCEAR 1977 estimate.

131. (Page E.9, Table E.4) The column titled "Mays(19)" may contain an error. Mays, *et al.* (reference 19) estimated 200 bone cancer deaths/10⁶ person-rads. Using a Q of 10, this would be equivalent to 20 bone cancer deaths/10⁶ person-rem. In a contemporary paper (Mays, C.W., Estimated Risk from ²³⁹Pu to Human Bone, Liver and Lung, pp. 373-384 in Biological and Environmental Effects of Low-Level Radiation, Vol. II, IAEA, Vienna, 1976), Mays estimated (again assuming a Q of 10) 20 lung cancer deaths, 20 bone cancer deaths and 10 liver cancer deaths per 10⁶ person-rem.

EPA in its guidance on transuranium elements (EPA 520/4-77-016) provided an analysis of the health impact of exposure to transuranium elements in the environment which includes both risk and dose-rate estimates for a cohort of 100,000 exposed since birth. This guidance is supplemented by technical reports, Technical Report EPA 520/4-78-010 and Technical Note CSD-78-1, which provide background information for the basic guidance document. Since the health impact calculated in these reports is based on lifetime exposure and risk coefficients for specific organs, the results are not directly comparable with Table E.4 but they are a more realistic estimate of health impact from transuranium elements in the environment.

132. (Page E.10, paragraph 6) Although BEIR, 1972 did not provide a risk estimate for skin cancer, the 1978 Stockholm meeting of ICRP suggested if a skin cancer risk is required, an estimate of 1 fatal cancer per 10⁶ person rem could be used. Averaging the risk estimates in UNSCEAR, 1977, the skin cancer incidence is around 0.5 cases per year per 10⁶ person-rem; with a 5 percent mortality this would be about 2 fatal skin cancers per 10⁶ person-rem. The 1977 UNSCEAR Report suggests alpha risk might be higher.

133. (Appendix F) The description of the geology at the bottom of page F.2 would be improved if there were some indication of the depth of the basement rocks and of the general nature of the overlying rocks.

If the maximum flood of record was 3 meters above the normal river stage, and the one in a thousand year flood would be expected to be 5 meters above normal river stage, under what conditions would the "maximum probable flood" which is 10 meters above normal river stage be expected? Is this a once in a million year flood?

There is a short paragraph on ground water, but nothing as to the nature of the aquifer--permeability, hydraulic gradients, or retardation factors. This should be included.

134. (Appendix H) This appendix could well be omitted. Many of the hazard indices quoted are of no value as indices, and there is no information to enable one to select which, if any, of the indices are useful.

Although purporting to be a basis for determining the "hazard index," the material as presented in the appendix does not even approximate the potential hazard. The MPC is derived on the basis of dose to a "critical organ" rather than on the risk related to a given intake of isotope. The cumulative risk from intake of isotopes should be used as the basis for deriving a comparative "hazard index" since organ sensitivities are the controlling factor as noted in ICRP 26.

135. (Appendix I) This appendix is deficient. It is based on leaching of the entire repository by ground water, passage of the nuclides through a rather freely flowing aquifer, and discharge into a large surface stream (10,000 cubic feet per second or 8.9×10^{12} liters per year). If we apply the generic density of population in terms of river flow from our forthcoming dose assessment report, which is 3.3×10^{-7} person years per liter, the river is capable of being a water supply for about 3 million people, a great many of whom would receive close to the maximum individual dose.

There is an apparent conflict between the basic assumptions in the main text and Appendix I. The main text stated that "...disposal in salt has been emphasized..." (Page 3.1). However, the assumptions made in Appendix I (Page I-9) for an earlier analysis (which was the basis of the current version of the impact statement) assumed that the repository is in a non-salt media. Furthermore, some of the details of the model should be briefly summarized in the appendix. The statement, "Detailed descriptions of these models are found in references 1-7" (Page I.4), is not sufficient. There should be a brief description of GETOUT (Page I.10), as well.

The consequence analysis as it bears on geological repositories appears to be too subjective. For example, the draft continually relies on a holdup period of about 1,000 years. Yet an independent study (EPA/520/4-78-004) by an Ad Hoc Panel of Earth Scientists concludes that "The containment cannot be relied upon to effect any significant (longer than 10 years) retardation of the release of the high-level waste." While it is obvious that finite numbers must be used as input for the programs employed in this appendix, the large variation possible in much of the input data suggests that incorporation of an uncertainty analysis and sensitivity analysis in such calculations would permit a clearer understanding of possible consequences.

The criterion for public acceptability of 120 millirems per year to the maximum individual is not defensible, and population dose needs primary consideration. If the approximately 3 million people who could be supported by the river were each to receive 120 millirems per year, the population dose would be 350,000 person-rems per year or approximately 70 health effects per year using the BEIR-I whole body estimate. Although all the postulated 3 million people would not receive the maximum individual dose and although these nuclides would not produce whole body doses, there is no reason to believe that the concentration of nuclides in the river would decrease substantially as the nuclides moved down river.

The analysis uses unquoted sorption equilibrium constants. These are probably the Battelle desert soil values which may be unreasonably high. The text on page I.6 refers to "5 miles of western U.S. subsoil," which is reminiscent of the Battelle "desert soil." These sorption constants are not necessarily typical of all soils and rocks and, in any case, should be listed in tabular form.

Just as it is improper to neglect population dose in the river, it is improper to neglect individual dose to users of ground water. This is completely omitted in this section. Since the ground water velocity is stated to be equal to one foot per day, or a little over 100 meters per year, the aquifer would be expected to be a good water provider and

comparable to the aquifer of Appendix F which is stated as supporting "numerous shallow wells supplying residences and farms" and also a "public water supply well" for A City. Population dose from use of the aquifer may very well be significant in addition to individual doses.

The discussion of compensating for a poor site by an extremely durable waste container in the last paragraph of page I.2 is irrelevant, since human intrusion cannot be ruled out.

The concentration on individual dose rather than population dose is again shown in the fourth paragraph on page I.3 which speaks of reducing the iodine-129 dose by a factor of 10 by reducing the release rate by a factor of 10. Population dose would not be changed.

136. (Figure I.2) Why are there zero's on a logarithmic plot?

137. (Page I.7) The leach rate figures used throughout and specifically in Figure I.3 are unrealistically low. The "hypothetical waste management system characterization" is about a factor of ten better than the values we have been given by our consultant, Arthur D. Little, Inc., and contrast strongly with the estimate of the EPA geologist panel: "There is no evidence that incorporation into a glass will ensure resistance to significant leaching over time scales over a decade." (page 7, EPA 520/4-78-004).

138. Figure I.4 appears to require a leach time of 100,000 years for "satisfactory" (less than 120 millirems per year) operation. This may not be possible for all contained nuclides, since some nuclides are geochemically mobile.

139. (Page I.10) The notion that the dose from Ra-226 can be reduced by limiting the leaching of U-238 is incorrect. It is doubtful that U-238 migration could be controlled over its half-life (4.5 billion years).

We believe that the impact analysis is in a premature stage in this section. The analyses stated in Appendix I are divided into two categories: past work and present work. Since the present work is only partially complete, the results presented in the DEIS may be revised when the present work is completed. This may change results in the stated conclusions in the DEIS. We believe the present work should include an error analysis and sensitivity analysis.

All the references to this appendix are from Battelle Pacific Northwest Laboratories work. Has any of this work been performed elsewhere?

140. (Appendix J) Figure J.1 should be explained. Its applicability is unclear.

141. (Appendix L) The statement that devitrified glass is stronger than ordinary glass and will resist further fracturing is not as important as the potential greater leaching from devitrified glass.

In this Appendix, and in Appendix M, there appears to be no consideration of any accidental releases other than sabotage.

142. (Appendix M) The accidents leading to releases of radionuclides (Tables M.3 and M.8) are not characterized, so it is impossible to understand what is involved. The basis for release of 0.1 percent of total krypton-85 (page M.52) is not given. The total releases of 22 Megacuries of krypton-85 should be compared with the permissible 40 CFR 190 values. There is no consideration of possible radionuclide releases from accidents in a spent fuel storage facility in Table M.52. There is some discussion in Table M.61 but there is no basis for judgment as to the releases or selection of accidents. For example, there is no discussion of the effect of loss of coolant in water basin storage through failure of the tank or through sabotage.

Note also that the risk estimates, pp. M.6, M.33, M.53, M.81, M.87, etc., will require revision if numerical risk coefficients are changed since all are derived from the risk coefficients developed in Appendix E.

143. (Appendix N) Estimated costs are given for transportation of spent fuel and waste, but there is no indication as to how these costs were found. The bases for these estimates should be presented or referenced in the Final EIS.

144. (Page N.13) "Doses to the maximum individual...and population dose are comparable." This statement does not make sense, since there is a 10,000 times difference between the maximum individual dose and population dose in Table N.12. This should be clarified in the Final EIS.

145. (Appendix O) The 1,000 year storage and surveillance assumptions used in the calculations are in conflict with proposed Criteria for Radioactive Wastes (43 F.R. 53262 *et seq.*, November 15, 1978) developed by EPA. The appendix should be revised using the proposed period of storage and surveillance of no more than 100 years.

146. (Appendix P) Ringwood and co-workers have identified a suite of minerals for use in waste disposal. Their work should be referenced and seriously considered.

147. (Appendix Q) This is a rather interesting appendix although the development of the field does not appear to be sufficiently advanced for any convincing environmental impact analysis. There seems to be a contradiction between Tables Q.4 and Q.5. In Table Q.4 a 1 meter barrier is reported to retain strontium-90 and cesium-137 for about 30 years, or about one-half life for these nuclides. In Table Q.5 a 1 meter barrier is said to retain them for 30 half lives.

The possible competition for ion exchange sites on added minerals (or natural minerals for that matter) should be noted. Canister materials are elements of the transition series, notably iron, nickel, chromium, or titanium. In Sweden, lead and copper have been suggested for canisters. The ion exchange capacity of any added materials must be enough to handle the nonradioelements as well as the radioelements.

148. (Appendix R) This is an interesting discussion but the state of development of the technology does not permit more than qualitative information.



United States Department of the Interior

OFFICE OF THE SECRETARY
WASHINGTON, D.C. 20240

ER 79/397

NOV 14 1979

Honorable Ruth Clusen
Assistant Secretary for Environment
Department of Energy
Washington, D.C. 20545

Dear Ms. Clusen:

Thank you for your letter of April 19, 1979, transmitting copies of the Department of Energy's draft environmental impact statement for the Management of Commercially Generated Radioactive Waste. Our comments are presented according to the format of the statement, or by subject.

General

From a NEPA and FLPMA viewpoint, we believe the draft statement will have to be substantially revised and supplemented in order for this Department to make use of it in support of any future land withdrawal decisions we may wish to make. Thus, our specific concerns are discussed in each of the sections below with a view that the final statement will take into account our concerns, especially those concerns under FLPMA. We will be pleased to work with you in the revision of the final statement to the extent that we have the capability to do so.

We believe the principal conclusion in the draft statement, that our technology is much further advanced in conventional geological disposal than in any other viable option, is stated as a premise to justify the cursory, qualitative treatment given all the other options relative to geologic disposal. The attention given to the other nine options has not ensured a genuine comparison of advantages and disadvantages. The statement implies that it is not really valid to compare conventional geologic disposal with the other options at this time because inadequate knowledge is available on the other options. Nevertheless, the statement proceeds to make these comparisons.



We recognize that a trade-off takes place between the degree of detail in analyses and the comprehensiveness of the documentation. Sometimes less detail is necessary to gain understanding. In this statement, the latter course might have proved useful. The constant use of technical jargon seriously weakens the analysis regardless of its scientific merits. This is especially true with regard to the discussion of potential design failures related to the transport of nuclear waste across the country to a repository.

One important issue, the possibility of deliberate reopening of a radioactive waste disposal site, should be presented in the final statement. Such sites should be secure and guarded over the long-term to prevent future release of radioactivity.

Relationship to the IRG Report

The President has recognized the immediate and long-term problems of nuclear waste management. In March 1978 he established a Federal Interagency Review Group (IRG) for nuclear waste management. The task force released its recommendations in a draft report in March 1979. This report also contained IRG's responses to the extensive public comment on the subject.

The IRG's summary statement appears to be more candid than this EIS. On page 42 of the final report (TID-29442) the IRG stated:

"Present scientific and technological knowledge is adequate to identify potential repository sites for further investigation. No scientific or technical reason is known that would prevent identifying a site that is suitable for a repository provided that the systems view is utilized rigorously to evaluate the suitability of sites and designs, and in minimizing the influence of future human activities. A suitable site is one at which a repository would meet predetermined criteria and which would provide a high degree of assurance that radioactive waste can be successfully isolated from the biosphere for periods of thousands of years. For periods beyond a few thousand years, our capability to assess the performance of the repository diminishes and the degree of assurance is therefore reduced. The feasibility of safely disposing of high-level waste in mined repositories can only be assessed on the basis of specific investigations at and determinations of suitability of particular sites. Information obtained at each successive step of site selection and repository development will permit reevaluation of risks, uncertainties, and

the ability of the site and repository to meet regulatory standards. Such reevaluations would lead either to abandonment of the site or a decision to proceed to the next step. Reliance on conservative engineering practices and multiple independent barriers can reduce some risks and compensate for some uncertainties. However, even at the time of decommissioning, some uncertainty about repository performance will still exist. Thus, in addition to technical evaluation, a societal judgment that considers the level of risk and the associated uncertainty will be necessary." (Emphasis added)

A discussion of these issues should be included in section 3.1.6, Research and Development Needs, in the final statement.

Technological Issues

The statement is an impressive encyclopedic compilation of information on radwaste generation and disposal. It presents the state-of-the-art for high-level and transuranic waste-disposal technology and is markedly superior to earlier documents. The major conclusion that disposal of radwastes in mined repositories is, for the near-term, the preferred approach is in keeping with conclusions reached by the Interagency Review Group (TID-29442).

However, the statement appears biased in its technological optimism. Unlike the IRG's measured optimism regarding the feasibility of geologic disposal of radwaste, this document leaves the reader with the impression that all existing uncertainties are solvable, given time and money. While we agree that radwaste can, with rigorous application of the systems approach, and with generated implementation procedures, be safely isolated from the biosphere for a few thousand years, this report implies that such an outcome will be the normal result of existing and as-yet-to-be-developed technology. The final statement should recognize that:

- a) Radwaste disposal is a new technology which, like all preceding technologies, will develop over the course of decades in response to experiences which in all likelihood could include some failures;
- b) Successful design of systems in the geotechnical fields differs markedly from that in other engineering fields, as succinctly stated recently by C. H. Dowding: "The process of exploring to characterize or define small-scale properties

of substrata at construction sites is unique to geotechnical engineering. In other engineering disciplines, material properties are specified during design, or before construction or manufacture, and then controlled to meet the specification. Unfortunately, subsurface properties cannot be specified; they must be deduced through exploration" (1979, Site characterization and exploration: Amer. Soc. Civil Engr., p. 1). Similarly, prediction of the synergistic effects of geologic, hydrologic, and geochemical processes on a radwaste repository over millenia is not a straightforward modeling process. As much has been admitted by H. C. Burkholder whose pioneering modeling studies form the basis for the EIS's optimistic conclusions that repositories pose no significant risk to man. (See closing statement in Burkholder, H.E., et al., 1977, Safety assessment and geosphere transport methodology for geologic isolation of nuclear waste materials, in Risk analysis and geologic modeling in relation to disposal of radioactive wastes into geological formations: Proc. of workshop organized jointly by OECD Nuclear Energy Agency and the Commission of European Communities, Ispra, Italy, p. 216-229);

- c) Catastrophic failures have recently occurred even in well-advanced technologies, for example, Apollo 6, Apollo 13; the collision of two 747's on Tenerife Island (Canary Islands), Teton Dam, and the DC-10 pylon issue.

Site Selection

Of the 10 alternative methods described in the EIS for disposal of nuclear wastes, the first one--geologic disposal using conventional mining techniques--appears to have the most potential for impacting lands administered by our Bureau of Land Management (BLM) in Oregon and Washington. The Columbia River basalt geologic formation, which covers substantial portions of northeastern Oregon and southwestern Washington, is one of the formations given prime considerations for disposal sites under this alternative. Substantial acreage of BLM-administered public lands are located on this basaltic formation. Because this programmatic statement is completely non-site-specific, we wish to draw your attention to potential impacts on public lands should nuclear waste facilities be located there. The final statement should respond to the following information even though it is not possible to identify site-specific conflicts at this time.

An important criterion for suitable geologic host formations is that they have not been extensively drilled, mined, or altered by the hand of man. This is also a prime characteristic for existing and potential wilderness areas. BLM is reviewing public lands for potential wilderness values under Section 603 of the Federal Land Policy and Management Act of 1976 (43 U.S.C. 1782), and Section 2(c) of the Wilderness Act of 1964 (16 U.S.C. 1131). BLM's wilderness review process must be taken into consideration in any discussion of the environmental impacts associated with alternative geologic formations considered for nuclear waste disposal sites.

Our Bureau's greatest concern is that the site selection, characterization, and evaluation process will involve many potential locations on public lands with subsequent application for withdrawals for future sites which might affect the wilderness selection process. These issues must be addressed in order to comply with FLPMA. The final statement should recognize that a potential conflict with the designation of wilderness areas may occur for specific sites on public lands. In addition, we strongly urge that you include sufficient detailed information in the final statement that would outline those requirements of FLPMA that pertain to withdrawal of public lands for a waste disposal site. This inclusion is necessary because hundreds of miles of public right-of-way in the West are bounded on either side by vast tracts of public land. Any release of nuclear waste could impact public lands and programs.

To enable Bureau of Land Management to meet NEPA requirements, the discussion should indicate that public lands throughout the West have uses which could be in conflict with a nuclear waste disposal facility, and the degree of impacts to public lands and resources would vary from site to site, and from region to region. The most significant impact would be on the long-term productivity of the affected environment as all uses not directly supporting the nuclear waste disposal facility would be eliminated from the thousands of acres necessary to secure the site. These above and below ground uses include, at a minimum: water resources, range land use for livestock, wildlife, and wild horses and burros; forestry; recreation, and cultural resources; wilderness and areas of critical environmental concern; oil and gas exploration, and extraction; nonenergy minerals, etc. In addition, since BLM's capabilities for multiple-use management of the public lands are prescribed by law, we recommend the trade-offs between storage of nuclear waste and existing uses of public lands be given more attention in the final statement.

The final statement should address the requirements for withdrawals of public lands for any alternative involving storage of nuclear waste in geologic formations, or the Outer Continental Shelf. Environmental hazards or conflicts of Outer Continental Shelf storage, with regard to Outer Continental Shelf oil and gas exploration and production, need to be addressed. The issue of irreversible loss of resources at any site should also be included as the result of preemption of an area for geologic disposal of radioactive wastes. In particular, salt domes may contain oil, gas, sulfur, potash, or other commercial minerals. Exploratory drilling of the proposed site would be the most desirable means for obtaining the subsurface data necessary for determining the location and extent of possible economic mineral deposits. However, we recognize that any such drilling program would adversely affect the geologic integrity of the site, and evaluation of its mineral potential must, therefore, be based on other sources of information that may be available.

Lastly, the draft statement tends to avoid discussion or use of site-selection, evaluation, and qualification criteria, apparently because NRC has not issued formal criteria yet. However, there are general published criteria available (such as those of the National Academy of Science and the U.S. Environmental Protection Agency) which might cover the generally agreed-upon major issues. Criteria could be used to more effectively present the advantages, disadvantages, and unresolved technical, sociological, political, and esthetic issues involved with the various disposal options.

Land Use and Transportation Considerations

The final statement should delineate willingness to minimize environmental impacts which may be precipitated by the proposed action on the Nation's cultural, natural and recreation resources. Therefore, the final statement should address statutory environmental requirements, e.g., the National Historic Preservation Act, as amended; Section 6(f) for the Land and Water Conservation Act, as amended; provisions of the Instrument of Transfer for surplus property; and Executive Order 11593, which further the Federal Government's policy to preserve, restore and maintain the historic and cultural environment. The final GEIS should include clear, coherent identification and analysis of the environmental impacts which may be reasonably expected to disturb or affect the Nation's cultural and natural and recreation resources. This Department's Heritage Conservation and Recreation Service would be pleased to provide technical assistance in this subject area upon request.

The final statement should stress that the interim storage, permanent storage, and transport of commercial radioactive wastes will be carried out in a manner that has no potential for adversely affecting units of the National Park System, the Wild and Scenic River System, and the National Trail System. Further, as trustee for Indian trust lands, we require that Indian communities (1) be fully aware of the permanent hazards and potential dangers the placement of such materials will have upon them, (2) the effect upon their future generations, and (3) their full support and agreement for emplacing the radiological materials. At present we favor the exclusion of such activities from areas which could affect Indian trust lands.

The final GEIS should indicate that such areas would be specifically precluded at the program level from consideration as potential disposal sites, and would be bypassed in all transportation operations.

Ecosystems

Above-ground industrial-type facilities could occupy lands used for production of forage for various animal species. This would be a relatively small area and, depending on the specific site, would probably have minimum impacts on the total vegetation and soil resource. However, there might be significant impacts in the surrounding buffer zone of 10,000 to 20,000 acres.

Since significant and varied impacts on ecosystems are potentially associated with the management of commercially generated radioactive waste, the final statement should at least identify what these potential impacts are rather than indicating in section 4.5.8 on page 4.22 that there is not sufficient information to allow impact evaluation. Such an addition would ensure the fullest possible disclosure of impacts and, thus, strengthen the final statement.

Appendix S describes the Ecosystem Impact criterion for assessment of the impacts of alternatives. The discussion is limited, however, to pre-emption of ecologically productive land and does not relate to the short- or long-term effects on the soils, plants, or animals occupying the potential sites, nor of adjacent off-site lands. The criterion should be expanded to consider the radiological effects on plants and animals in addition to humans.

Modeling of Groundwater Migration

We believe the comprehensive model used in the safety analysis is not applicable on a generic basis. The modeling efforts of H.C. Burkholder and his colleagues at Battelle are pioneering and commendable. However, in Appendix I the assumptions used in the model analysis are clearly spelled out on page I.9. Among these assumptions are: a) that "the repository is located in a non-salt formation surrounded by a geology with nuclide retention properties similar to those for a particular Hanford Reservation subsoil;" and b) "the ground water flows into a surface stream with a flow rate of 10,000 ft³/sec (1/10 the flow rate of the Columbia River near the Hanford Reservation) where the nuclides are further diluted." This flow is equivalent to the average flow of the Delaware River at Trenton. With these and other simplifying assumptions, the model predicts a benign outcome. However, the problems are multiple.

First, although dilution of the radionuclide-bearing ground water by a 10,000 ft³/sec river is one plausible scenario for radwaste dissolved in Hanford ground waters, a 10,000-fold concentration might occur in other environments, for example, in areas where ground water flow is toward marshes or wet playas. Second, what is the dose to man if the ground water were tapped by a future town well-field upgradient from discharge into the river? Third, the Kd's for Hanford subsoil are unlikely to be applicable to fractured media.

Briefly, the model is acceptable for one HLW scenario in Hanford alluvium. It is unacceptable for other scenarios at Hanford, and certainly unacceptable for any other rocks and waste types. It follows that the seemingly comprehensive tables comparing health effects from radwaste disposal in salt, granite, shale, and basalt are difficult to justify. The draft EIS itself in several places follows the IRG in emphasizing the importance of site-specific studies. Therefore, we suggest the presentation of considerable numerical data in Section 3.1.5.2 is not warranted; this should be resolved in the final statement.

Ground-water transport

If a systems approach is to be used in siting and engineering mined repositories, we believe the consequence analysis should consist of a systematic consideration of failure of each element in the system. It has been stated many times (e.g., IRG Subgroup, 1978, TID-288/8, app. A, p. 16) that transport by moving ground water is the most likely means by which toxic radionuclides may reach the biosphere. The multiple barrier approach is designed to avert this eventuality. Yet, the consequence analysis in the EIS treats this possibility in less depth than four other "worst case" scenarios--meteorite impact, diversion of a surface or underground river into the repository, drilling, and solution mining

of salt. Emphasis on these unlikely release mechanisms seems unbalanced and could cause undue apprehension as to the risks involved. Ground-water transport is treated at greater length in appendix I but this material clearly should be up front. Even appendix I does not analyze for variation of key parameters such as retardation, porosity, and permeability.

There are additional reasons for laying more stress on release by moving ground water in the main part of the statement. Although this is a generic statement and values for hydrologic parameters are site-specific, the possible ranges of these parameters are relatively well known. A credible consequence analysis would, therefore, show the effects of varying porosity, permeability, hydraulic head, path length, retardation, release rate, etc., over reasonable ranges. For environments in granite, basalt, and shale relatively rapid flow through fractures should be included in the analysis. One of the key parameters to be considered is retardation. The present analysis uses values for the Hanford subsoil--highly site-specific and uncertain, inasmuch as values determined by various laboratories continue to differ by significant amounts. An analysis that shows the effects of a range of retardation values is therefore especially critical.

Another reason for stressing variation in hydrologic parameters in the consequence analysis is that, while the ranges of these are known, the probabilities of initiating events (with the exception of meteorite impact) are much more uncertain. It will be argued below that the probability used for faulting is unsupported; and the probabilities assigned to human activities sometime in the distant future are generally conceded to be meaningless (IRG Subgroup, 1978, TID-28818, app. A, p. 50, 52). The most likely result of human intrusion (aside from serious effects to a few individuals) is release to ground water, again emphasizing the need for analysis for all barriers to nuclide migration.

There is a danger that, like the consequence analysis performed by EPA in its standard-setting procedure, the attempt to be "conservative" will lead to acceptance of repository site characteristics that violate the multiple barrier approach. The ground-water transport analysis in the main body of the statement uses a path length of only 10 km, apparently in an attempt to show that consequences would not be drastic. The inter-agency effort to find acceptable sites now going forward will certainly not consider sites with such a short path. The analysis should use a longer flow path for the base cases and discuss consequences of shortening of the path due to tectonic and/or climatic change.

C.45

The release rates used in the ground-water transport analysis in the main body of the report jump strangely from 100%/yr to 0.1%/yr. The base case should clearly be for the low rates, but results for intermediate release rates should be presented. The consequence analysis should consider solubility limits of the various radionuclides in the ground-water system under consideration; the present analysis assumes varying source terms for nuclide transport, some of which may not be physically possible.

Long-term surface storage

Long-term (30-60 year) surface storage of radwaste should be considered in the final statement as a means of decreasing uncertainties of geologic disposal. The final statement should discuss the utilization of extended (30-60 year) surface storage of HLW prior to disposal as a means of reducing major rock mechanics uncertainties created by the heat pulse. Such storage would reduce the heat pulse by one-half to three-quarters. The omission of this alternative is surprising in light of the extended discussion in section 3.2 of chemical synthesis which, similarly, is a way to reduce geochemical uncertainty in geologic disposal. We recommend the final statement consider long-term surface storage as a viable alternative.

Multiple barriers

The multiple barrier concept and the systems approach to mined repositories, elaborated in the IRG reports, do not occupy the central role envisioned by the IRG in this EIS. Words that include these concepts appear in the statement, but they are obviously last-minute additions and include major errors in places. For example, on page 3.1.1, "multiple barriers" are listed as one of six characteristics of conventional geologic disposal when, in fact, three of the other five characteristics are themselves important barriers to nuclide transport. Apparently, what is meant by multiple barriers here is only the hydrologic systems beyond the host rock. However, the desired properties of the hydrologic system are not necessarily the same as those of the disposal medium. Indeed, since predictability of the system is important, it might be advantageous to have part of the hydrologic flow system include a porous medium in which transport is relatively well understood as opposed to a relatively impervious medium subject to flow-through fractures, which is much less well understood.

Clearly, the draft EIS has not given sufficient thought to the total system of containment, but considers its components separately.

Radiological safety

The draft statement employs a consequence analysis; events with the worst possible outcomes are postulated and the subsequent radiological health effects described. These effects are considerable for some events, but the discussion tempers this outcome by introducing the concept of risk. Serious consequences, it is argued, have a low probability of occurring so the net risk (probability x consequence) is judged to be qualitatively small.

It should be noted that the risk need not be zero and will not be zero for any waste-management option or energy system. Also, it is not necessary, for the purposes of this statement, that a complete risk analysis be presented--only that enough be understood about the risk to justify continuing with the option. This GEIS can be considered adequate when viewed in this light. It makes clear that the earth has a potentially high retentive capacity and that properly sited and engineered repositories should lead to a relatively low risk. Whether this risk will be acceptable (i.e., judged safe) is for society to decide.

Faulting

The value of 4×10^{-11} /yr for the probability of faulting or fracturing (Claiborne, H.D., and Gera, F., 1974, Potential containment failure mechanisms and their consequences at a radioactive waste repository in bedded salt in New Mexico: Oak Ridge National Laboratory, ORNL-TM-4639) used in risk considerations is outdated and its uncritical acceptance is a major shortcoming of the draft EIS. This is not to say that the values for faulting or fracturing probabilities ultimately used for a site-specific risk assessment will not be some low number such as this, but these probabilities will have to be determined on a sound basis. Research to do this should be identified in the section on research and development needs.

Claiborne and Gera (1974) assumed that faulting would be random within a given region and that the rate of fault initiation for the Delaware Basin could be approximated by assuming a constant rate over post-Permian time. Both assumptions are highly unlikely in the light of recent tectonic thinking. It is now realized that most tectonic strain is taken up on existing faults as long as the tectonic regime remains the same. When the tectonic regime changes, existing faults will continue to take up much of the strain; but new faults may form depending on the new stress state and its relation to existing discontinuities. The new faults probably form over a relatively short time as the new tectonic regime is established.

Reasonable bounds for these rates can be formulated with data now in hand from the geologic record, but a research effort to do so should be put in place.

The discussion in the EIS uses the probabilities for faulting and fracturing interchangeably, but they will not necessarily be the same. Clairborne and Gera (1974) treated the problem of major fault development across the repository. Fracturing could be induced by distant seismic events or by the presence of the repository itself. Whether fractures can remain open in salt long enough for transport to take place remains an open question that needs emphasis in the EIS.

Values for southeast New Mexico should not be used in a generic statement. Certainly, faulting rates of 10^2 /yr, typical of the San Andreas system, could be avoided. A reasonable, "conservative" upper bound for this analysis might be 10^{-4} /yr. The discussion on page 3.1.156 would then conclude that the risk from repository breach by faulting and flooding would be no greater than the risk from lightning, assuming the rest of the analysis is correct--not definitely seven orders of magnitude less.

Another possible effect of geologic storage of radioactive wastes is the increase in underground pressure as a result of entrapment of gases (helium, radon, etc.) released through radioactive decay schemes. This increased pressure, if not properly relieved, could lead to the development or reopening of fissures that would result in the escape of radioactive materials to the surface. The final statement should address this potential event.

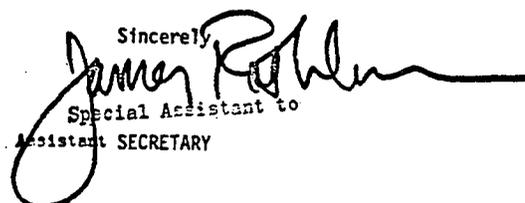
Decision grid

The use of the decision grid approach may be premature and unjustified at this stage. It invites invalid, quantitative comparisons of options to be made. Even though the report warns against this, there will always be those readers who tend to seize the numbers and use them improperly. Other concerns with the decision grid approach are also expressed in the attachment.

Other options

Other options such as seabed disposal and deep-well disposal also have to be considered promising for the future and should be pursued simultaneously with conventional geologic disposal. If properly pursued, one of the other options may eventually prove to be superior to conventional mined repositories, in which case it should be adopted. This need for continued pursuit of other technologies should be more heavily stressed in the final statement.

We hope these comments will be helpful to you in the preparation of a final statement.

Sincerely,

Special Assistant to
Assistant SECRETARY

Attachment

SPECIFIC COMMENTS

The title of the report appears somewhat inappropriate because low-level wastes and mill tailings, which are major components of commercially generated wastes, are not addressed. Therefore, this title should specify "high-level and transuranic radioactive wastes" instead of its more general connotation.

Page 111, line 5: Change "reasonable" to "greatest" or "high."

Page 1.1, par. 3: It certainly has not been established that mined geologic repositories are the best option--only that they may be an acceptable option for the first phase of disposal. Therefore, (2) does not follow from (1). Proposal (2) implies that work on promising options other than conventional mined repositories will be dropped. Perhaps insertion of the word "first" before "...operative disposal technology" will help.

Page 1.12. Second sentence under Media Properties. This sentence is not correct because rock falls account for a large number of injuries and fatalities in the mines.

Page 1.27 (2nd paragraph). If there is sufficient heat to modify the red clay of the ocean floor, there may be sufficient heat to initiate convection currents in the overlying water. If sufficiently large in areal extent, this would cause an upwelling, bringing to the surface material from the lower depths of the ocean and possibly from the ocean floor. This material could be nutrients, inert material, or if a canister ruptured on impact, radioactive material.

Page 1.33. Fuel cycle compatibility is left out of the table.

Page 1.3.7. Leaving the waste in a solar orbit or disposing of it on the moon still leaves the waste available for future exposure to man. Would not direct injection into the sun be preferable?

Page 1.5, par. 1, line 10: This paragraph implies that foreign spent-fuel or reactor wastes are not currently being sent to this country. We are under the impression that some foreign spent fuel is coming into the country now. Is this true?

Page 1.5, par. 3, line 1: Insert "should" after "systems."

Page 1.5, par. 4, line 7: A type of canister should depend primarily on disposal medium, waste form, and buffer-backfill conditions. Suggest striking "and exposure" and substituting "waste form and other..." The word "Absorptive" should be "Adsorptive."

Page 1.5, par. 4, line 10: Change "absorptive" to "adsorptive." "(V)ery deep hard rock formations" is not a specific medium and should be deleted.

Page 1.6, par. 1: It appears that too much emphasis is being put on institutional controls. Although works of many ancient civilizations are evident today, many others have been obliterated and lost. There is still much speculation on the purpose of many Inca and Mayan structures, not to mention Stonehenge. We have still failed to locate many dwellings, cities, and other work known to have been in existence somewhere. It is also clear that many ancient works have been the object of vandalism, destruction, probing and plundering (i.e., Great Pyramids). It is not reasonable to expect repository markers to remain evident for millennia. In that time period, they could easily become stolen, destroyed, or covered by sedimentation or other works of man.

Page 1.6, par. 2: This conclusion has not yet been established by evidence in the report and is out of place.

Page 3.1.7. Next to the last sentence. This sentence implies that the only difficulty with temperature is fracturing the overlying rock. Whether the overlying rock could be fractured in this manner would depend on the depth of burial, as well as rock properties, and it might be expected as a second order effect of shallow repository site. The first order effect of temperature would be the spalling and other failures of the cavity skin.

Page 3.1.10. In the last paragraph it is indicated that igneous rocks closely related to granite might not be suitable because of trace element and mineralogic composition. However, the "Sierra Nevada granites" are shown in figure 3.1.2. These are predominantly quartz monzonites and would this eliminate them from consideration?

Page 3.1.13. Third paragraph. In discussing basalts and jointing, they have left out the fact that basalts are layered with discontinuities, volcanic ash, "soil," sandstones, etc., between the layers. The zone between layers ranges in permeability from opened to sealed.

Pages 3.1.1 to 3.1.8. The geothermal temperature gradient has been left out of the discussion on geologic siting considerations. At 500 meters depth, it would probably not be important. However, at depths greater than 1,000 meters, the natural rock temperatures will be high enough to consider in the heat flow analyses. Also, the rock temperature comes into any calculations that are made for ventilation requirements during construction and operation of the facility.

Page 3.1.28. Fifth paragraph, last sentence, is not correct. More people are injured and killed from falling rocks in coal and noncoal mines than are injured by rock bursts. The shale environment that is considered as one of the potential repository sites will have problems of ground support similar to those in coal mines.

Page 3.1.30. Third paragraph, first sentence. Timber is more common than steel for use in sets with lagging. This sentence should be changed to read ". . . to (timber or steel sets with lagging)."

Third paragraph, second sentence. This sentence is not completely correct. It should be modified to read ". . . drilled into the rock and either mechanically anchored or grouted to the rock."

Third paragraph, fourth line. The sentence starting on this line should read: "Steel wire mesh, metal sheets, or beams are. . . ."

Third paragraph, seventh line. The sentence starting on this line should read "Timber or steel sets are structural. . . ."

Page 3.1.30. Footnote. Comment on shale strength. Most shales are layered and their properties perpendicular to the layering are much different from those parallel to the layers. Hence, any number on shale properties should reference its orientation.

Page 3.1.31. Second paragraph, fourth line. The statement "(sometimes called rock bursting)" should be removed because it propagates confusion on the true meaning of rock burst. It is not necessary to the sense of the sentence which will read, with its removal, "Slabbing during construction can sometimes be sudden and hazardous."

Page 3.1.35. Last paragraph. The resins used with resin grouted bolts are polyester, not epoxy.

Page 3.1.36. Paragraph on seismic loads. Third line in that paragraph. Remove the word "explosions" relative to rock burst, and let the sentence read ". . . seismicity (earthquakes), rock bursts, and other phenomena."

Page 3.1.123. Fourth paragraph. Nonradiological Accidents. A common fatal/nonfatal accident rate was used for surface construction activities which seems reasonable. However, a common rate was also used for the underground construction. This rate was derived from underground mining other than coal. To be closer to the truth in this area, the accident rates for more representative industries should be used. For the salt repository, the accident statistics from salt mining and potash mining should be used. For granite and basalt, underground metal and nonmetal hardrock mining is more appropriate, and for shale, use the coal mine accident statistics.

Page 3.1.137. Line 7. This line contains the first mention of phosphates. What is the tie-in with salt repositories that might be constructed?

Page 3.1.147. Second line from bottom, typo on sentence starting "Such scenarios. . . ."

Page 3.1.178. Fifth line from the bottom, typographical error on involves.

Page 4.27. Third dot item. Has ISF been defined prior to this use? The ISF, intermediate scale facility, is defined on page 4.34.

Page 5.8. First paragraph, last line. ". . . 3) use of nuclear parts as a safeguard measure. Should parts be parks?

ISF is not listed in the glossary.

Page 1.6, par. 3, sentence 2: It should be pointed out that we know little of the actual geochemical and hydrologic conditions prevailing during the Oklo nuclear reactions, making it difficult to relate those phenomena to any present environments; nor do we know much of the history of conditions which prevailed in the millenia after the reactions were sustained.

Page 1.12, par. 4: There is too much emphasis on the 600 m depth here and in other places in the report. That is an apparently arbitrary depth. The actual optimum depth is dependent not only on rock type but geohydrologic conditions and other site and waste characteristics.

Page 1.12, par. 5: The four media mentioned are not "representative of all geologic media." They are only examples of the types being considered for repositories; suggest rewording to that effect.

Page 1.13, par. 3, last sentence: Not all of shale's properties are "intermediate"--perhaps only its thermo-mechanical and hydraulic properties. Its adsorptive properties are probably superior to the other three media and its hydraulic properties could be comparable or superior in some cases.

Page 1.13, par. 4, line 10: Not only hydraulic gradients should be measured in situ but other hydraulic and geochemical parameters such as permeability, porosity, and sorptive properties.

Page 1.13, last par.: This is a gross understatement of the current deficiencies in the data base. Specific gaps in rock-property knowledge should be itemized. The interaction between waste and host rock is a major area of deficiency and thus should not be passed off matter-of-factly. Other major unresolved issues should be mentioned: shaft and borehole sealing technology; development of adequate nondestructive in situ geophysical techniques; demonstrations of adequate large-scale, in situ measurements of solute transport phenomena (characterizations of sorption and other reactions and dispersion coefficients, especially for far-field analyses); quantitative methods to analyze and model ground-water flow and solute transport in fractured rocks; development of adequate method to measure hydraulic properties of very-low-permeability rocks and fractured rocks; ability to predict geologic and climatic processes.

Page 1.14, par. 1, last sentence: This is an overstatement; suggest striking "assure" and substituting "will be used to provide high assurance."

Page 1.14, par. 2: The data base for granite is rapidly approaching that for salt, thanks to the recent work in Sweden, Canada, U.K., and U.S. We know considerably more about granite than basalt or shale.

This paragraph again understates the deficiency in the geologic data base. It stresses media deficiencies and fails to mention inadequate technology

in nondestructive, in situ testing and other problems mentioned in our comments for the previous page. The problems of predictive geology are not adequately expressed in the final sentence of the paragraph. There are at least as many deficiencies in the geologic data base as in "Waste Form;" yet "Waste Form" is allotted five times as much space as "Geology."

Page 1.14, par. 4, lines 7-9: These two sentences indicate an attempt to dismiss a real problem. Presumably, water will eventually enter any mined repository, except perhaps one above a deep water table in an arid western basin. Leachability is still an unresolved issue.

Page 1.15, par. 3: The proposed Swedish canister is not "highly" sophisticated; it is a simple copper can with lead fill. The engineered sorption barriers are not part of the canister but part of the backfill buffer around the canisters. In the last sentence it is not clear what "redox materials" are.

Page 1.16, par. 2, line 11: Many toxic organics will degrade biologically or chemically with time; we therefore suggest inserting "might" before "remain."

Page 1.16, table 1.3: This table could be misleading, in the sense of comparing apples to oranges. Chlorine gas, for instance, will rapidly deteriorate in most environments because of its high reactivity. Phosgene and ammonia are also non-persistent. Most of these substances can easily be treated to render them relatively harmless.

Page 1.17, par. 1: The difference between a "major disaster" and a "primary event" is not clear.

Page 1.17, par. 2, lines 6-7: We disagree; uncertainty of geologic predictions does limit the application of risk assessment. If the probability of a certain geologic event occurring is not known, how can a reliable risk assessment be calculated to include the potential impact of such an event?

Page 1.18, par. 5: It would seem that different disposal options and media would involve different canister and buffer materials. This might have a major impact on certain materials such as stainless steel, copper, and bentonite.

Page 1.21, par. 3: The basis for calculating the frequency of stream invasion of a repository is not clear. Appropriate references should be cited.

Page 1.22, par. 1: It is not "unreasonably pessimistic" to assume that such markers would be stolen, moved, maliciously destroyed, covered by natural sedimentation, or other works of man.

Page 1.23, par. 2: The material used for the canister might also be valuable enough in the future to attract invasion of the repository. Copper, stainless steel, titanium, and even gold have been mentioned by some as canister material.

Page 1.23, par. 3: Institutional controls cannot be relied upon for more than several decades.

Page 1.23, last par. line 2: Why 600 m?

Page 1.23, last par. line 4: Suggest inserting "chemical" before "thermodynamic."

Page 1.23, last par. line 5-6: Nothing is absolutely chemically inert in any ground water, so it is unrealistic to suggest that the waste form might be so rendered.

Page 1.23, last par. line 8: Suggest substituting "approaching" for "in" before "thermodynamics."

Page 1.24, par. 3: The economic feasibility of the synthetic-mineral waste form must also be assessed.

Page 1.23, last par. last line: Suggest substituting "might" for "would."

Page 1.25, par. 2-3: There are major problems with this option, if the hole is lost (collapses or is otherwise rendered unuseable) during the waste emplacement or backfilling-sealing stages. We would end up with the waste in the wrong place or irretrievably placed in an unsealed hole, both of which are probably unacceptable. These potential problems should be pointed out.

Page 1.26, par. 2: Other unresolved problems or disadvantages that should be mentioned regarding island disposal are: greater probability of disruptive geologic events (faulting, earthquakes, volcanism, etc.); perhaps greater erosion rates; more subject to higher rainfall rates, tropical storms, and tsunamis.

Page 1.26, par. 7, line 2: Suggest substituting "appear to be" for "are" before "abyssal."

Page 1.26, last par.: Suggest inserting "sub-seabed" before "geologic media."

Page 1.27, par. 2, line 4: Suggest substituting "possible" for "the" before "thermal-related."

Pages 1.28 and 1.32: The discussion and assessment of the reverse well injection alternative should include at least general consideration and evaluation of the vertical separation of individual plates or sheets of injected waste-bearing grout. Consideration of the impacts of the alternative should

also include at least generalized assessment of effects of possible deviation of sheets from bedding planes and possible unplanned or unscheduled accumulation of wastes in any given zone.

Page 1.28, par. 5, last line: Suggest inserting "chemical" before "precipitation."

Page 1.28, par. 6, line 2: Suggest inserting "most probably" before "be by ground water."

Page 1.28, last par.: Although grout injection technology is rather well established, long-term durability-reliability of grout seals is unknown. It appears that considerable effort is needed to develop grout types and sealing techniques that can be relied upon with confidence for many millenia.

Page 1.29, par. 2: It should also be stated that radioactive-waste disposal of any kind is prohibited or severely restricted in several states.

Page 1.29, par. 3, line 3: Suggest substituting "isotopes" for "wastes."

Page 1.34, table 1.8: As pointed out in the "General Comments" above, we have major reservations with the use of this type of approach in this EIS. First of all, not all experts would agree that the most important criteria and attributes have been included. For instance, a category for foreign policy conflicts is included, but not one for internal political controversy or State-local conflicts. This would be significant in the case of sub-seabed disposal which suffers from a low-rating in international conflicts, but would enjoy a high rating in a category for State political controversy.

We are of the opinion that the data base for the criterion of long-term radiological safety is insufficient to make national rankings in all options and all attributes at this time. It is inconsistent to make estimates (which are subjective value judgments) for some criteria and not others, such as "Socioeconomic Impact," "Aesthetic Impact," and "Ecosystem Impact."

Perhaps it would be better for this exercise to be based on a more widely accepted, or independently generated, set of criteria such as those of the National Academy of Sciences, EPA, or IAEA.

Many of the rankings are really value judgments by the "panel of experts." In section 4.1 and appendix S, it is emphatically stated that the matrix approach was used to minimize value judgments. Yet most of the rankings in this matrix can be considered value judgments. Therefore, the table is highly subject to influence of the bias and prejudices of the "experts." According to appendix S, all 14 members of the panel of experts are associated with prime DOE contract organizations. This means they could be suspect of bias toward DOE preferences because they are undoubtedly working on some aspect of radioactive waste disposal (directly or indirectly) that

is financed by DOE. For example, there probably are people at each of those institutions working on some aspect of conventional geologic repositories. If the panel were comprised of these investigators, the results would probably be biased toward that option. The panel should be comprised of a significant number of experts outside the pro-nuclear industrial-government sector (NAS, NRC, EPA, USGS, academic institutions, environmental groups, State agencies, etc.).

The very fact that one option, which has been studied considerably, is being compared to others with much weaker data bases would tend to bias the procedure immediately. Most experts would tend to rank the better understood options higher than the lesser understood options, even if the latter has more promise.

We would disagree with several of the numerical rankings given in the table (again reflecting value judgments and bias).

Page 1.36, lines 1-2: Just because a factor is a good discriminator, does not mean it is necessarily an important one. The fact that some attributes showed little discrimination could mean they were misjudged.

Page 2.1.16 and 2.1.18: It was noted in the Department of Energy's recent draft environmental statement for the Waste Isolation Pilot Plant in Eddy County, New Mexico, that drilling into the stored spent fuel 100 years after burial could expose the geologist on the drilling crew to a whole-body dose of about 90 rems, which is 18 times the annual occupational exposure that is now considered permissible (p. 1-7 in WIPP statement). It would be useful to show graphically for comparative purposes how such exposure levels would decline over the first several hundred years and at what point they reached levels closely comparable to exposures that would result from drilling into various uranium ore deposits. Possibly such a graph would help dispel public concern for the long-term fate of buried radioactive waste. If a date can be established at which the two consequences are considered truly comparable, it would be useful to show that date on graphs and tables whenever appropriate, or alternatively to show the levels of radioactivity considered truly comparable to natural ore deposits. For example, it would be particularly useful to show such levels of radioactivity on such tables as 2.1.9 and 2.1.11. Under the uranium and plutonium recycle option it is noted that the total radioactivity of actinides and daughters actually increases slightly from 100,000 to 1 million years after waste disposal (table A.3B), but if the level is below that of typical uranium ore deposits, it would appear to be of little significance.

Page 2.1.24: It would be helpful to provide a readily understandable summary of the energy values present in the uranium, plutonium, and thorium in the radioactive waste, and particularly in the spent fuel, in order to clarify what would be thrown away under the three fuel-cycle options. There are

brief references to the fact that deferred disposal or reprocessing of spent fuel would maintain "future options to recover the energy values represented by the uranium and plutonium contained in the spent fuel elements" and would "permit a more informed decision on the advisability of reprocessing the spent fuel" (par. 2). However, we found no quantification of what those energy values actually are, and we believe that in order to promote a more informed decision on those questions it would be helpful to include a clear appraisal of the amounts of potential energy that would be wasted by irretrievable disposal, possibly expressed in units comparable to barrels of oil or tons of coal. It might also be useful to show these values of potential energy obtainable under various fuel cycles including uranium recycle, uranium-plutonium recycle, various breeder fuel cycles, and thorium-based cycles.

Page 2.2.2, item (2): The formula given for allowable whole body radiation dose should be checked for possible error. It is given as $5(N-18)$, N being age. For a 48-year-old, this appears to give a dose of 150 rems, suggesting that it might be a lifetime dose limit. In any case, the meaning is unclear.

Pages 2.1.1 to 3.1.46: Sections 3.1.1 and 3.1.2 in particular are redundant and contain some errors of fact, significant omissions, and numerous juxtapositions of unrelated subject matter. These chapters appear to have been put together by culling material from a variety of sources. The length of these chapters can be considerably reduced and their value increased many fold if subjected to detailed review by earth science editors.

Page 3.1.1, par. 1: Why not cite published, well accepted general geologic criteria such as those of NAS? Criterion No. 6 should indicate that a long, slow flow path is one of the barriers. Low rainfall (aridity) is an additional desirable feature.

Page 3.1.1, last par., last 2 lines: Suggest inserting "topography and rock properties" after "Climate;" substitute comma for "and" after "wind", and insert "and chemical thermodynamics" after "gravity."

Page 3.1.2, par. 1: The validity and methods of relating paleoclimatic conditions to future variations and erosion rates has not been established. This area requires more research.

Page 3.1.3, par. 2, last line: Suggest inserting "and rock properties" at the end of the sentence, because they, too, control erosion rates.

Page 3.1.3, par. 4: It should be noted that these depositional processes can obliterate repository location markers.

Page 3.1.4, par. 2: There are no completely insoluble radioactive compounds (or other compounds).

Page 3.1.4, par. 3: Porosity is just as important as permeability in determining rates of ground-water flow. This should be so stated.

Page 3.1.5, par. 1, last sentence: This sentence is not necessarily true; future tectonism may not be reflected in past tectonic history. This factor will probably have to be a probability-based estimate.

Page 3.1.5, par. 2: The terms "permeability" and "hydraulic conductivity" are somewhat different in meaning and should be defined. (They are not in the glossary.)

Page 3.1.5, par. 2, line 5: Suggest substituting "dissolve" for "assimilate."

Page 3.1.5, par. 3, line 2: Suggest inserting "chemical precipitation reactions" after "radioactive decay."

Page 3.1.5, par. 3, sentence 2: The term "convection" is misused here. Convection in ground-water flow means mass flow along any head gradient. Thermal convection would imply convection induced by temperature gradients.

Page 3.1.5, par. 3, line 4: Suggest substituting "through" for "into" after "ions."

Page 3.1.5, par. 3, last sentence: "Sorpton" processes do not generally include chemical precipitation and filtration processes.

Page, 3.1.5, par. 3: We suggest that the appropriate definition of dispersion should be included with the definitions given for major mechanisms related to nuclide transport. Dispersion may be especially important in hydrogeologic environments where diffusion may be too slow to be significant in the distribution of contaminants, e.g., where fracture permeability is a factor or where relatively high hydraulic gradient and low recharge may be controlling influences.

Page 3.1.5, par. 4, line 1: Suggest inserting "quantities" after "types."

Page 3.1.5, last par. line 9: Suggest inserting ", 3) and most importantly the rate of flow" after "flow."

Page 3.1.6, par. 1: It should be pointed out that fractures and joints may also be important permeability features of shale.

Page 3.1.6, par. 2, lines 6-8: Core analyses are not satisfactory for permeability of the medium, especially if it is fractured.

Page 3.1.6, par. 2, last sentence: Field testing techniques are not yet developed to make adequate determinations for very low permeability or sparsely fractured media.

Page 3.1.6, par. 3, line 2: A difference in "hydraulic head," not pressure, is necessary to induce ground-water flow. Suggest substituting "hydraulic head (pressure plus gravitational component)" for "pressure (hydraulic gradient)" after "difference in."

Has the possible significance of a density gradient been considered among the inducements to fluid movement?

Page 3.1.6, par 3, line 7: Suggest inserting "generally" after "medium is." This entire section on Hydrology of Host Rock (p. 3.1.5-6) needs more emphasis on ground-water velocity, porosity, dispersion, and fracture-flow problems.

Page 3.1.6, par. 6, last line: It should be mentioned that fuel elements or other high Pu-content wastes might be considered a desirable resource in the future.

Page 3.1.6, par. 7, line 3: The multiple barriers should not "act together" but independently, so that if some fail the others compensate.

Page 3.1.6, lines 4-5: The multiple barrier components include: waste form, canister, buffer material, length of ground-water flow path (far-field), adsorption and other reactions. These should be added to the sentence.

Page 3.1.7, par. 2, line 7: Substitute comma for "and" after "strength" and insert ", chemical (sorptive)" after "thermal."

Page 3.1.7, last par., line 1: Substitute "conduct" for "dissipate" and insert "away" after "heat."

Page 3.1.8, par. 1: It should be mentioned that degree of anisotropy and heterogeneity are also important considerations.

Page 3.1.8, par. 3, line 7: Insert "and high permeability" after "porosity."

Page 3.1.9, table 3.1.1: Basalt or granite do not always or even usually have a permeability of "n₁." Shale minimum permeability is much lower than 10⁻⁴, perhaps 10⁻¹¹ or 10⁻¹² ft/yr. Moreover, the key hydrogeologic parameter for evaluating these rocks, as repository hosts, is transmissivity, not permeability. Yet, this parameter is omitted.

Page 3.1.9, par. 1, lines 4-7: The plasticity of salt can also be an undesirable feature if diapirism is reactivated in a salt dome or initiated in a bedded deposit.

Page 3.1.9, par. 1: The assertion is made that water incorporated in salt beds when the beds were formed does not migrate. This should be qualified by reference to effects of elevated temperatures on migration of brine. Also, in the last paragraph on this page, it should be mentioned that one of the problems with salt is that brine contained within the deposit tends to move toward heat sources, such as radioactive waste. These hot brines can be highly corrosive to some canister materials and waste forms.

Page 3.1.10, par. 1, line 2: Insert "generally" after "It is" and "principally" after "composed."

Page 3.1.10, par. 1, line 3: Insert "fairly" after "generally."

Page 3.1.10, par. 1, line 7: Substitute "very deeply rooted" for "relatively bottomless."

Pages 3.3.11-3.3.12: It appears that in the context of the duration of periods under consideration in repository planning, the discussion of rock strength needs to include at least in a general way changes in rock strength and characteristics that may occur with increasing time and their effects on permeability.

Page 3.1.11, last par.: Some mention should be made of sorption characteristics of granites. The significant recent work on granite at the Stripa Mine and other areas of Sweden, should be cited, as well as British work on granite. The Canadians have also made significant studies recently, which should be cited.

Page 3.1.12, par. 2, line 4: The sentence "Voids are ..." is inaccurate. Suggest substituting "because of the extremely fine pore size of shales, intergranular permeability is generally very low. However, fractures and joints in some shales substantially increase bulk permeability."

Page 3.1.12, par. 4: The main advantages and disadvantages of shales should be summarized in a final paragraph.

Page 3.1.13, par. 1, line 1: Insert "Columbia River" after "Generally." Many other basalt flows do not have large areal extents.

Page 3.1.13, par. 1, line 3: Delete portion of sentence after "because of" and substitute "their high permeability."

Page 3.1.13, par. 2, line 1: Insert "commonly" after "basalt is."

Page 3.1.13, par. 2, line 1: Again, the omission of mention of the transmissivity of basalt is inexcusable. The statement on the low permeability of basalt is a half-truth, misleading to a reader without hydrogeologic training.

Page 3.1.13, par. 3, end: Insert "Joints are unfavorable because of their relatively high permeability, low surface area, and potentially high ground-water flow rates."

Page 3.1.13, par. 4, item 2: The category of "general crystalline rocks" would also include basalts, which are listed separately.

Page 3.1.14, par. 2 and table: It should be stated who made the ratings, since they are value judgments. We believe the table is misleading and oversimplistic. Any of the media listed can have a wide range of properties

indicated, ranging from excellent to poor. High porosity is not necessarily undesirable because, for equal permeability, a rock with high porosity will have slower ground-water flow rates than one with low porosity. Thermal conductivity of basalt is much greater than shale; hence the numbering given is backwards. Again, no mention is made of the transmissivity of the media considered even though this parameter, unlike permeability, would permit a separation of media on a scale of 1-3 (salt would rate 3, granite 2, shale 2, and basalt 1). Other important properties were left off the table. These might include:

Plastic properties

Sorption properties

Page 3.1.16, par. 3: It is stated that "High-level radioactive wastes must be kept isolated from the biosphere for a long time period, perhaps hundreds of thousands of years," but previously it had been stated that "The HLW must be managed...to isolate the uranium, plutonium, the balance of the TRUs, and some fission products for several millennia, and probably up to one million years" (p. 1.9, par. 2, last 4 lines). It would be desirable to state these time periods consistently and to add any necessary qualifications such as may relate to the form of the waste.

In order to avoid unwarranted public concern it is imperative to avoid any appearance of a reluctance to acknowledge the full time period that the wastes will be potentially of appreciable hazard to man. Since any inconsistency in describing the duration of the potential hazard tends to give this appearance, we feel that all references to the required duration of isolation should be completely consistent, carefully qualified (by reference to waste form, disposal method, fuel cycle option, hazard level, etc.), and should relate the radioactivity levels or potential hazards to familiar natural sources of radioactivity whenever possible (such as uranium ores, monazite sands, cosmic radiation at high altitudes, etc.). Some comparisons presented in the EIS appear to deserve more emphasis in order to dispel unwarranted concern and put the potential hazard of radioactive waste into proper perspective. For example, the total toxicity of plutonium projected to be wasted in the year 2000 is comparable to the toxicity of lead wasted in 1973, but the plutonium will gradually decay while the lead will persist indefinitely (p. 3.1.65, next-to-last par.). If the required period of isolation of the radioactive waste actually proves to be a full 1 million years, this period of time should not in itself cause great concern if the waste were either in a stable form or well isolated; concern for the 1-million-year duration might be dispelled by comparison with natural phenomena, such as the several-billion-year age of the earth, the fact that the earth's crust contains an entire group of rocks more than 2.5 billion years old, and the fact that

1 million years is the amount of time required for our spiralling galaxy to rotate through less than 2 degrees of arc (based on one rotation in approximately 200 million years).

Page 3.1.16, par. 6, line 3: Substitute "should be capable of being adequately" for "can be quite well" after "of an area."

Page 3.1.17, last par.: Who should set the criteria?

Page 3.1.18, par. 2, line 3: Whose criteria?

Page 3.1.20, par. 1, line 4: Substitute "is injected into" for "injects" after "mobile care."

Page 3.1.20, par. 1, line 14: Insert "tectonic" after "the."

Page 3.1.21, fig. 3.1.8: Stage II data base should also include regional geophysics survey (resistivity and seismic profiles, magnetics, and gravity).

Page 3.1.22, par. 1, line 6: Insert "It appears that" before "Some aspects of."

Page 3.1.22, par. 1, line 7: Substitute "might" for "can" after "million years."

Page 3.1.22, end of par.: Insert "The driving mechanisms for plate tectonics are presently not understood."

Page 3.1.23, par. 3, line 6: Substitute "low flow velocities" for "low hydraulic gradients."

Page 3.1.23, par. 4: Some mention needs to be made that considerable work is needed in the fields of measuring and modeling fracture-flow hydrology, measuring very low permeability, and other field-scale or in situ measurement of hydraulic and solute transport parameters.

Page 3.1.23, par. 6, last line: Strike "impermeable;" no rocks are impermeable.

Page 3.1.26, par. 3, line 2: The reference to "thousands of years" should be to "probably up to a million years" or to "hundreds of thousands of years" for consistency with statements on pages 1.9 and 3.1.51.

Page 3.1.26, p24. 3, lines 10-13: The question of predicting future resource value is understated. How can we know what will be valuable 1,000 years or more from now? Who would have guessed 200 years ago that uranium or even petroleum would be a valuable resource?

Page 3.1.26-28: The section on Deficiencies in Data Base is much too general and non-specific. For example, it fails to mention lack of data on long-term

shaft and borehole sealing; large-scale sorption measurements; long-term verification of models; large-scale dispersivity measurements; and other deficiencies.

Page 3.1.27, par. 2, line 5: The large increase in volume of gypsum from anhydrite could cause detrimental stresses in a repository; this should be mentioned.

Page 3.1.28, par. 1: Because of the great public concern over hazards of radioactive waste from commercial nuclear power reactors, we believe that considerable effort would be justified toward the objective of translating levels of radioactivity of the waste into terms that are readily understandable to the public, and that facilitate a comparison with naturally radioactive materials or with natural sources of radioactivity. In the present statement, considerable effort appears to have been made in accomplishing this type of comparison in the area of routine and accidental exposures to radioactivity from artificial sources such as radioactive waste by comparison with natural sources such as the earth, the cosmos, and the human body. However, it would be helpful to make similar comparisons between the radioactivity of the disposed waste (for example, as permanently buried in a geologic repository) and the radioactivity of various typical uranium ore deposits, at various times after disposal of the waste. The brief discussion here (p. 3.1.28) is not in sufficient detail to be useful in evaluating probable consequences of drilling into buried waste canisters at various future dates by comparison with drilling into typical uranium ore deposits.

Page 3.1.29, par. 3, line 4: Insert "and hydraulic head gradient" after "mass."

Pages 3.1.29, par. 3, last 4 lines: The inflow of such high quantities of ground water would probably disqualify the site.

Page 3.1.29, last par., last line: After last sentence insert "However, there would be wide variation in these spacings from place to place."

Page 3.1.32, par. 2, line 2: Insert "per unit head gradient" after "unit time."

Page 3.1.29, par. 2, lines 6-9: It seems highly unlikely that criteria would allow a repository in an area with such high ground-water inflow.

Page 3.1.29, par. 4: At end of paragraph, insert "However, studies in Sweden, Great Britain, and Canada indicate that considerable fracturing can be found in granites at postulated repository depths."

Page 3.1.29, par. 6, lines 3-4: The questions of shaft and borehole plugging apply to all media, not just salt.

Page 3.1.35, par. 1: It should be mentioned that heat sources in salt draw water toward them.

Page 3.1.39, par. 3, line 3: Substitute minimal for "no" after "transport pathways."

Page 3.1.39, par. 3, line 4: Substitute "increase flow of" for "introduce flowing" after "changes could."

Page 3.1.39, par. 3, line 5: Insert "and through" after "ground water to." Substitute "should" for "will" after "geologic medium."

Page 3.1.41, par. 5: Interception of an aquifer would certainly violate repository criteria and would preclude the option of backfilling emplaced waste.

Page 3.1.44: Somewhere in section 3.1 a discussion should be included on the present problems and uncertainties of long-term borehole, shaft, and tunnel backfill and sealing technology. If the shafts and disturbed zones around them are not adequately sealed for the time required, they would act as a dangerous short circuit from the repository to the biosphere.

Page 3.1.47, par. 3, lines 2-3: Hydrology should also be a prime consideration in geologic site selection.

Page 3.1.47, last par: The possible value of spent fuel as a resource should also be mentioned for consideration.

Page 3.1.48, par. 1, line 5: Insert "and to provide as long a pathway to the surface as feasible" after "surface phenomena."

Page 3.1.48, last par., last sentence: This sentence is an overstatement. We do not have adequate, accepted techniques for testing fractured rocks or rocks with extremely low permeabilities or high degrees of anisotropy and heterogeneity. We are especially lacking in nondestructive field techniques (those which do not require a lot of wells directly in the repository areas).

Page 3.1.49, par. 2, sentence 1: Add transmissivity, storage coefficients, boundary conditions, and porosity to list of factors needed.

Page 3.1.49, par. 2, line 3: Substitute "available" for "one of several" after "From."

Page 3.1.49, par. 3, line 1: Substitute "very low" for "low to no" after "The areas of." Every place has some ground water.

Page 3.1.49, par. 3, lines 6-8: Laboratory methods are not satisfactory for measurement of in situ hydraulic parameters in any type of media (especially fractured). This section should be deleted.

Page 3.1.49, par. 5, line 2: Insert "even in salt or very tight clayey shale" after "the repository." Delete the following sentence.

Page 3.1.50, par. 1: Mention should be made in this paragraph that satisfactory borehole sealing techniques (for very long periods) have not yet been developed.

Page 3.1.50, par. 2, lines 3-5: This is an overstatement: these techniques are not "well in hand." Nondestructive testing techniques, for example, are not adequately developed.

Page 3.1.51, par. 5: Again, this is an oversimplified, over-confident statement, without technical basis.

Page 3.1.50, par. 7: The two sentences in this paragraph are contradictory and constitute over-simplified rationalizations.

Page 3.1.52, par. no. 3, last line: "Sorption characteristics" should be inserted after "rock units."

Page 3.1.53, par. 6, line 7: Change "silica" to "silicate."

Page 3.1.63, last par., line 4: It should be noted also that man invaded the Egypt "repositories" (pyramids) before he knew what the hieroglyphics said.

Page 3.1.64, par. 1, line 1: There are many more examples of institutional and political systems that have not survived more than a few centuries, than have survived.

Page 3.1. 64, par. 5, last line: Insert "United Kingdom" after "France."

Page 3.1.64, par. 7, line 4-5: It is stated that "after several hundred years of decay, the wastes do not exceed the natural radioactivity of the ores from which they came" (p. 3.1.64). This statement is confusing, because it gives the impression that drilling into an underground waste repository after several hundred years would have consequences no more severe than drilling into the ore deposits from which the uranium was mined. If that were strictly true, it should be more clearly explained whether, and how, the wastes differ in their potential hazard from natural uranium ores after several hundred years, and to explain in a manner clearly understandable by the public why it is considered that high-level wastes must be managed for isolation for "probably up to one million years" (p. 1.9). If the comparability with natural uranium ores after several hundred years is not strictly valid, insofar as potential hazard is concerned, it would be advisable to explain significant differences.

Page 3.1.65, Hazard Indices: This section contains several specious arguments culled from various sources. Either the arguments should be presented in detail or the whole section should be eliminated. In fact, the closing paragraph of the section on page 3.1.66 itself suggests that the preceding page is devoid of meaning.

Page 3.1.65, par. 4: See criticisms of same table on page 1.16.

Page 3.1.67, par. 2, line 4: ^{229}Ra should be ^{226}Ra .

Page 3.1.68, Lithosphere/Biosphere Transport: While the reader is correctly advised that "Some ground water and transport models have been calibrated," he is not told that modeling of flow through fractured aquifers is in its infancy.

Page 3.1.68, par. 5, second sentence: This is not true for fractured media.

Page 3.1.68, par. 5, line 8: An additional, more relevant model calibration was done by J. B. Robertson (1974, Digital modeling of radioactive and chemical waste transport in the Snake River Plain Aquifer at the National Reactor Testing Station, Idaho: U.S. Geological Survey Open-file Report (AEC-22054)).

Page 3.1.70, par. 3: The last sentence in this paragraph contradicts the first two sentences.

Page 3.1.71, Consequence Analysis: It should be pointed out that all numerical models will require more satisfactory verification on a variety of real field problems before they can confidently be applied to very long-term and large-scale predictions.

Page 3.1.76, par. 2: There is a brief reference to the issue that "We should delay implementing geologic isolation until we are more certain that the wastes have no practical value now or in the future." Nowhere in the EIS have we found any discussion of the pros and cons of that issue, or the pertinent data to support an informed decision. Because of increasing scarcity of energy sources and other natural resources, it would appear that the statement should at least briefly assess the very-long-range impacts of ultimate and irretrievable disposal of commercially generated radioactive waste on the basis of any conceivable future utility to the human environment.

A related matter, that could only be discussed in general terms prior to selection of tentative disposal sites, is the potential impact on recovery of leasable or other minerals that may be found at a disposal site.

Page 3.1.83, par. 1, lines 6-7: It is stated that "Required activities are described for four possible fuel cycles," while previously it was stated "Three fuel cycle alternatives are considered" (p. 1.7, par. 4).

Page 3.1.106, par. 3: What will the rooms, shafts, and tunnels be back-filled with? They can potentially serve as ready avenues for ground-water flow.

Page 3.1.114, par. 5: There can be potential problems from placing low-level wastes next to high-level wastes if the low-level waste has significant organic constituents or other chemical incompatibilities.

Page 3.1.123, par. 1: It does not seem credible for water inflow through shale to be ten times that through granite; granite is not necessarily less permeable than shale as table 3.3.1 (p. 3.3.7) points out.

Page 3.1.149, table 3.1.35: What is the source of the listed leach rate (1×10^{-4} gm/cm²/day) for intermediate-level and low-level wastes? What waste form(s) is (are) involved?

Page 3.1.225, par. 4: See our comments on the same subject for page 1.6 (par. 1).

Page 3.1.238, par. 5, line 4: Delete "core" after "require."

Page 3.1.238, par. 6: It should also be pointed out here that satisfactory shaft and tunnel sealing techniques have not yet been developed.

Page 3.1.239, par. 2, end: Add the sentence "However, such in situ tests have not yet been fully developed or proven."

Page 3.1.240, par. 3, last line: Insert "and sorptive properties" after "inter-crystalline fluids."

Page 3.1.241, par. 3, line 4: Insert sentence "By necessity, some foreign material such as cement or benetonite will be added to the backfill to make it more compact and less permeable" after "possible."

Page 3.1.243, par. 2: It should be pointed out here that sorption phenomena (or "Kds") are not yet well understood and characterized. Considerable fundamental and field research is needed in these areas for both near- and far-field analyses and modeling.

Page 3.1.243, par. 5, sentence 3: It is approaching late CY 1979 and the Presidential policy statement has not yet been made.

Page 3.2.1-3.2.23: Section 3.2, Geologic Emplacement Following Chemical Resynthesis, and the relevant appendix P represent, on the whole, original and innovative work on this promising concept. However, mineralogists and geochemists of the U.S. Geological Survey noted numerous misspellings of

mineral names and incorrect formulae, as well as some errors of fact and interpretation in these sections. Rather than detail these deficiencies here, we suggest that the authors of these sections set up a consultant group from universities and the USGS to review the concept and go over these sections thoroughly for accuracy and completeness. Although the concept is technically immature, the presentation in the EIS should make use of available expertise.

Page 3.3.3, par. 2: It should be reemphasized here that satisfactory backfilling-sealing techniques have not yet been developed and proven.

Page 3.3.7, par. 1, lines 3-4: In terms of fluid and solute migrate, fracture porosity might be the most important in many host rocks including basalt, granite and shale. Although there may be very few fractures, their permeability can be several orders of magnitude greater than that of the pores.

Page 3.3.7, par. 2, lines 7-8: Although fractures may be only a few meters long, they are often interconnected with others making a continuous flow network; therefore, the statement in the report is misleading.

Page 3.3.7, last par., lines 6-7: This sentence appears misleading or erroneous; oil companies have tested many wells below depths of 500 m for permeability. If the reference is only to crystalline rocks, that should be made clear.

Page 3.3.8: It should be pointed out here that permeability measurements for one well in a sparsely fractured medium have little transfer value to the surrounding bulk medium. Measurements on many wells drilled at different angles are needed (which might compromise the site) or some new nonpenetrating method is needed (not yet developed).

Page 3.3.29: The quantity "(1,000 ft.)" should read "(11,000 ft.)".

Page 3.3.37: Reference 23 appears to be incorrect. R. E. Goodman's article is entitled, "The deformability of joints." It is on pages 174 through 196 of ASTM Special Publication 477, published in 1970.

Pages 3.4.4 and 3.4.12: Probable ground-water migration and circulation patterns associated with the rock-melting alternative need further consideration and discussion, preferable in conjunction with effects of thermal cracking.

Page 3.5.16, par. 3: The reference to measurements in deep artesian wells seems inconsistent with the concept of a floating freshwater lens, because conditions in a confined or artesian aquifer might not necessarily reflect the freshwater/saltwater ratio. Influences other than differences in density can be effective in confined aquifers.

Page 3.5.21, par. 5: Another disadvantage should be mentioned. Islands are often more associated with resources and more subject to faulting, seismicity, volcanism, erosion, effects of sea-level changes, extreme storms, and tsunamis than mainland areas.

Page 3.6.6, table 3.6.2: The sorption coefficients listed are credible although they could easily be 1 or 2 orders of magnitude in error. Their source should be properly explained and cited. They should not be used for transport-time calculation.

Page 3.6.14, par. 1, last sentence: This statement requires documenting evidence or a reference.

Page 3.6.18, last par.: Appropriate references on these sorption coefficients need to be cited here.

Page 3.6.19, line 1: This statement appears in error; ^{54}Mn shows more than a factor of 10 spread.

Page 3.6.19, fig. 3.6.8: These sorption coefficients mean very little unless their exact measurement conditions (chemical and physical) are described. This would include as a minimum: ionic strength and composition of the solution, initial exchange conditions of the clay, pH and Eh of the solution, surface area of the clay, sorption equilibrium time, and solid-to-liquid ratio.

Page 3.6.19, last par., lines 4-5: Although organic complexes might be insignificant, there is a chance that carbonate (or other inorganic) complexing might be significant. This should be pointed out.

Page 3.6.29, par. 5-6: Measurements of molecular diffusion coefficients will be needed for near- and far-field transport analysis.

Page 3.7.10: Disposal of wastes in an ice sheet would probably entail loss of control of spacing of waste containers. Flow velocities and perhaps even flow directions might change over long periods. Possible impacts should be evaluated.

Page 3.7.10, par. 4: This statement should mention that there is one more handling and transportation step in ice disposal than in seabed disposal--from the unloading dock to the ice disposal site.

Page 3.7.10, sec. 3.7.1.6, first item: This statement is debatable; the fact that these remote areas are relatively unexplored for resources might attract considerable exploration in the future as some of the last frontiers for new discoveries.

Page 3.8.11, Candidate Geologic Environments: Effects of local stresses and conditions--such as varying topographic load and joint patterns--on uplift

resulting from reverse well injection should be further examined in the course of testing suitable monitoring methods that would not penetrate the individual waste sheets.

Page 3.8.24, sec. 3.8.3.1: Has reverse well injection in salt beds been considered? Salt is not mentioned among media under consideration. It has been found to be amenable to bedding plane fracture.

Section 3: It would be useful to provide concise summaries of advantages and disadvantages for the three subsections now lacking them: chemical resynthesis (sec. 3.2), very deep hole concept (sec. 3.3), and space disposal (sec. 3.10).

Section 4: See our comments for pages 1.31-1.35.

Page 4.2, par. 2: We believe that all the ratings used in this system require some degree of value judgment.

Page 4.25, par. 3: We believe that greater clarity could be achieved in expressing the several ideas included in the following sentence: "Further effort to improve knowledge of the performance of each CWM option with respect to the decision attributes and their parent criteria will also aid the decision makers in establishing important weighting factors needed to combine the attribute assessments into a composite figure of merit for each CWM option, for it has been demonstrated that the importance attached to an attribute or criterion depends not only upon the intrinsic nature of the criterion but also upon the expected performance of the decision alternatives with respect to that criterion."

Page 4.25, last par., last line: Add a sentence: "This does not imply, however, that conventional geologic disposal is environmentally the best option available."

Section 6, Glossary: Many important key terms are missing that should be included, such as: intermediate-level waste, permeability, hydraulic conductivity, Kd, sorption coefficient, distribution coefficient, beta activity, porosity, hydraulic head.



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

OCT 25 1979

Dr. Colin A. Heath
Director, Division of
Waste Isolation
U.S. Department of Energy
Washington, D.C. 20545

Dear Dr. Heath:

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the Department of Energy's Draft Environmental Impact Statement on the Management of Commercially Generated Radioactive Waste, DOE/EIS-0046-D, April 1979 (hereafter referred to as the GEIS). We have found many areas where modifications or additions to the statement are necessary. On the basis of our review, the staff offers the following general observations. Detailed comments on the GEIS are enclosed.

1. The conclusion drawn appears to be more comprehensive than can be supported.

The principal conclusion appears to be that "(1) the disposal of radioactive wastes in geologic formations can likely be developed and applied with minimal environmental consequences, and (2) therefore, the program emphasis should be on the establishment of mined repositories as the operative disposal technology" (GEIS, page 1.1, paragraph 3). However, information presented in the GEIS and its supporting documents does not appear to provide firm support for this comprehensive conclusion.

A number of critical areas have not been adequately dealt with. These include:

- long-term hydrogeologic transport of radionuclides from a geologic repository.
- effects on long-term repository isolation capability of repository construction and emplaced waste (thermal and radiological effects).
- potential effects of accidents during repository operation on the ability to properly backfill and seal the repository or safely remove the wastes already emplaced.

It may be that based on currently available information, such a comprehensive conclusion cannot be completely supported. If so, consideration should be given to restructuring the GEIS to support a more modest conclusion, perhaps similar to the following conclusion reached by the IRG (IRG report, page 42, para. 3).

Dr. Colin A. Heath

- 2 -

OCT 25 1979

"Present scientific and technological knowledge is adequate to identify potential repository sites for further investigation. No scientific or technical reason is known that would prevent identifying a site that is suitable for a repository provided that the systems view is utilized rigorously to evaluate the suitability of sites and designs, and in minimizing the influence of future human activities....The feasibility of safely disposing of high-level waste in mined repositories can only be assessed on the basis of specific investigations at and determinations of suitability of particular sites. Information obtained at each successive step of site selection and repository development will permit re-evaluation of risks, uncertainties, and the ability of the site and repository to meet regulatory standards."

This would provide support for a DOE program designed to proceed systematically to develop the most promising disposal options and as part of this program to proceed with the next logical step in development of mined geologic disposal, i.e., selection and characterization of potential sites in a variety of geologic media. We believe that DOE should proceed promptly with such development.

Such an approach explicitly recognizes the gaps in present knowledge and proposes a program designed to eliminate these gaps while proceeding toward development of an operative disposal technology.

2. Environmental comparison of alternative strategies for developing geologic repositories should be presented.

The Interagency Review Group on Waste Management (IRG) discusses several alternative strategies for developing geological repositories. The IRG left it up to the Department of Energy GEIS to do the full environmental analysis and comparisons of these strategies. The GEIS states that the various strategies will be assessed. However, the GEIS does not contain such an assessment.

The GEIS should examine each of the national strategies discussed (and any others deemed reasonable) in an explicit manner which permits an environmental comparison of the alternatives.

3. Environmental aspects of alternative timing strategies for commitment of waste to the repository should be examined.

In Sections 1.1 and 4.7.3, the GEIS concludes that the impacts of later implementation are insignificant. However, throughout the document, the implicit assumption exists that permanent disposal of the accumulated waste as soon as possible is an attribute of dominant importance.

C.59

OCT 25 1979

Although we agree that a repository should be developed and tested as soon as possible, it is not clear that there is a pressing need to rapidly commit existing inventories of high-level waste to the repository. A number of European countries, for example, are proposing long-term (40-50 year) surface storage of spent fuel or high-level waste prior to geologic emplacement. This is an important alternative to evaluate because (1) it provides ready retrievability should a reprocessing policy be adopted, and (2) the reduced thermal output of the waste will provide for either a smaller repository area requirement or a greater margin of safety with the same area. The environmental aspects of this alternative should be examined in the comparative assessment discussed in 2. above.

Therefore, we recommend that the environmental aspects of such delayed commitment of wastes to the repository be discussed in the final GEIS.

4. Comparison of alternatives is incomplete.

The abbreviated multi-attribute evaluation presented in chapters one and four is incomplete and of little value in comparing the alternatives presented in this report. The reported lack of sufficient data for comparison for several of the environmental factors and the absence of discriminative character of others has resulted in a comparison apparently based primarily on policy, rather than environmental considerations. This is inappropriate for an environmental impact statement.

Table 3.1.95 implies there is "no data" in a number of key areas essential to an analysis based on environmental considerations. If this table is correct, there is in fact no real environmental basis for comparison of the alternatives.

5. Decisions and decision processes should be identified.

The decisions and decision processes (i.e., who will make the decisions, how and on what schedule) which the GEIS is to support are not clearly identified in the GEIS. Such information should be included in the GEIS so that a reasonable assessment can be made whether the GEIS meets the requirements of NEPA.

6. The GEIS needs extensive technical and organization revision.

Our review has identified a number of apparent errors, over-simplifications, unsupported assertions, questionable assumptions, inconsistencies, and uses of outdated information in the GEIS. In addition, lack of proper documentation and referencing makes it difficult to check the technical accuracy of data presented. Although there is a wealth of valuable information in the GEIS and its back-up documents, information is difficult to locate and arguments difficult to follow.

OCT 25 1979

The GEIS and its supporting documents should represent the present state of knowledge concerning the disposal of long-lived radioactive wastes. Every effort must be made to assure the GEIS is technically sound in all areas, reflects the most up-to-date information available and is meticulously documented.*


John B. Martin, Director
Division of Waste Management

Enclosure:
As stated

*The Commission plans to conduct a rulemaking proceeding to assess its confidence that high-level radioactive waste can be safely disposed of. It is expected that DOE will be a principal party to this rulemaking proceeding and that the GEIS, if available in final form, will provide valuable input to this rulemaking proceeding.

COMMENTS ON
DRAFT ENVIRONMENTAL IMPACT STATEMENT
FOR THE
MANAGEMENT OF COMMERCIALY GENERATED RADIOACTIVE WASTES
(U.S. DEPARTMENT OF ENERGY, DOE/EIS-0046-D)

APRIL 1979

BY
THE STAFF OF THE
U.S. NUCLEAR REGULATORY COMMISSION
OCTOBER 1979

INDEX TO GEIS COMMENTS

	<u>Page</u>
1. GENERAL.....	1-1
2. FUEL CYCLE.....	2-1
a. Energy Projections.....	2-1
b. Waste Generation.....	2-1
c. Waste Storage and Treatment.....	2-3
d. Transportation.....	2-5
e. Safeguards.....	2-19
f. Other Fuel Cycle Alternatives.....	2-20
3. CONVENTIONAL GEOLOGIC DISPOSAL.....	3-1
a. Siting.....	3-1
b. Waste Form and Packaging.....	3-7
c. Design and Operation.....	3-7
d. Safeguards.....	3-17
e. Short-Term Environmental Impacts.....	3-19
f. Long-Term Effects of Repository Construction and Operation.....	3-23
g. Long-Term Radiological Effects - Environmental Transport.....	3-25
h. Long-Term Radiological Effects - Geology/Hydrology..	3-26
i. Long-Term Radiological Effects - Accident Analysis..	3-35
j. Research and Development.....	3-40
k. General.....	3-40
4. ALTERNATIVE DISPOSAL CONCEPTS.....	4-1
a. Geologic Emplacement Following Chemical Resynthesis.	4-1
b. Very Deep Hole Concept.....	4-1
c. The Rock Melting Concept.....	4-2
d. Island Disposal.....	4-4
e. The Sub-Seabed Geologic Disposal Concept.....	4-6
f. The Ice Sheet Disposal Concept.....	4-11
g. Reverse Well Disposal.....	4-12
h. Omitted Concept.....	4-12
5. COMPARISON OF ALTERNATIVES.....	5-1

C. 61

Comment
Number

1. GENERAL

1.1 The comparison of alternatives does not give sufficient consideration to environmental factors.

Table 4.5.1 indicates that insufficient data is available to compare ecosystem, aesthetic, and critical resource consumption impacts. These are among the most basic and fundamental, true environmental impacts. The majority of the remaining criteria are better described as policy considerations than as environmental factors, e.g., status of technology, cost of construction, policy and equity considerations. Thus, it appears that the final comparative analysis in this environmental impact statement drops out environmental factors and is based on the policy considerations. Environmental impacts, other than dose assessments, such as hydrologic impacts including water use and availability and impacts of construction and operation of the repository need more detailed discussion.

.2 The comparative analysis procedure is not carried to completion.

The GEIS is self contradictory on whether or not it is recommending a particular decision or decisions. In some sections it appears a certain course of action is being recommended. In particular on page 1.36, after eliminating most other factors as unimportant, it is stated, "Thus, state of technology stands out as a major decision factor, and the geologic disposal option has an edge over other options as regards the technology status." On page 1.1 it is stated: "DOE proposes that (1) disposal of radioactive wastes in geological formations can likely be developed and applied with minimum environmental consequences, and (2) therefore the program emphasis should be on the establishment of mined repositories as the operative disposal technology."

However, as indicated on page 1.31, the comparative analysis is intentionally not completed to "avoid value assumptions--more appropriately the responsibility of the decision maker." On page 1.35 is found: "It is

Comment
Number

emphasized that the scores in Table 1.8 cannot be combined without careful consideration of the relative importance of the attributes and of the criteria." The relative importance was not determined. Further, page 4.1 states that "No attempt is made to identify specific CWM options for further research and development." Page 4.24 reiterates that weighting factors have not been assigned and decisions not recommended.

The GEIS should not terminate the comparative analysis midway before assigning weighting factors, disclaim the making of a recommendation, and then proceed to make such recommendations as are found on pages 1.36 and 1.1. In deciding on which course of action to follow DOE should consider the CEQ regulations (40 CFR 1502.14 (e)) which require the identification of any preferred alternatives in the draft statement.

1.3 Time estimates for repository licensing schedules are too short.

The times allowed to complete licensing and construction of a repository are much shorter than those estimated by NRC. Figures 7.5.13 and 7.4.14 in DOE/ET-0028 show seven years from preliminary design to operation with one year between submission of a PSAR and construction approval. NRC estimates 10 to 12 years from preliminary design to a decision on operating approval. These longer times should be used in establishing repository availability dates as these delayed availability times may affect conclusions on the impacts of waiting until alternate methods are developed.

1.4 References to supporting materials are inadequately designated.

Although there is a wealth of information in the GEIS and its supporting documents, information is difficult to locate and arguments difficult to follow. References should be to specific page numbers in the supporting documents. It takes a substantial effort to find sources of GEIS information in the supporting documents. Many of the references are contractor reports. Where these reports relied on other sources, the prime reference should be given.

Comment
Number

- 1.5 The reference environment approach to generic evaluation is used improperly.

Inasmuch as the GEIS is a programmatic statement, a site-specific description of an environment is not necessary; however, development of data that will be required in a specific evaluation is appropriate, and the GEIS incorporates a reference environment to evaluate source terms on a generic basis. However once having determined the significance of an impact on the reference environment, the GEIS fails to remind the reader that conclusions reached relate only to those particular conditions. Indeed, statements in the GEIS indicate that even its writers do not fully appreciate these limitations. Effects on the reference environment are presented as the impacts of an alternative without recognition of the fact that the impacts could be much different for a different reference environment. For an example of how to prepare a GEIS with detailed discussions of siting options and impacts, see the FES on floating Nuclear Plants (NUREG-0056).

- 1.6 The use of a linear thermomechanical analysis for the design of the reference repository is salt is an inadequate approximation.

A conclusion that readily retrievable conditions in a repository in salt would prevail in storage rooms for at least five years is based on linear thermomechanical analyses. The time behavior of salt under thermal and mechanical loading cannot be approximated as a linear relationship. The GEIS makes an allowance of two feet to accommodate expected closures. Project Salt Vault, ORNL-4555, Chapter 12, presents several figures (Figure 12.36, etc.) that describe pillar behavior as a function of mechanical and thermal loading and time. The behavior exhibited is not linear and a significant underestimation of closure will be obtained if a linear approximation is used. If later studies have been used to discount the results of Project Salt Vault, they have not been identified in GEIS. Consideration of the creep behavior of salt under thermomechanical loading is important for the design of a repository in salt because it will affect the short- and long-term stability of storage rooms and access

Comment
Number

ways. If stability is compromised, the integrity of the impermeable barrier between the salt bed and overlying aquifers that is assumed by GEIS, may be compromised.

- 1.7 Retrievalability of the emplaced waste is expected to be a requirement during the early years of the conventional geologic repository. Based on various discussion within Section 3.1.2, it appears that many uncertainties and attendant much higher costs are introduced into almost every media as a result of waste thermal effects. Intermediate temporary storage of the waste for sufficient time to permit cooling to manageable temperature would seem to deserve considerable attention. The waste could, for example, be stored in large vaults at some minimal depth, then transferred to the final repository depth once acceptable temperatures had been attained. Perhaps either nonhigh-level waste could then be stored in the vacated temporary high-level storage area or the vaults could be filled with excavated rock from lower levels to minimize the surface environmental effects associated with disposal of some types of waste rock (such as salt).

- 1.8 The summary comparative analysis in Chapter 1 appears to be an attempt to justify conventional geologic disposal. Some of the alternatives still appear to have significant merit and, as indicated in the report, will receive further study by DOE. Conventional geologic disposal only happens to currently be at a more advanced stage than other technologies. The summary section should concentrate on the recommendations for the entire program, justifying each part of the continuing range of alternatives. This is a programmatic DOE decision and responsibility.

- 1.9 From the analysis presented it appears that nonradiological environmental impact considerations will not influence the selection among the six geological disposal options for a given fuel cycle option. However, even if this is true, consideration of environmental impacts will be important in site selection for any of the geological options selected. It is not readily evident whether one geological option should be selected before a

Comment
Number

comparison of alternative sites is made or whether, indeed, at a later date the site selection will include consideration of different geological options. Considerations of this type should be part of the "programmatic strategy" selection to be supported by the GEIS.

- 1.10 It seems inconsistent to identify some symbols, abbreviations and acronyms as footnotes and others in the glossary, e.g., Tables A-18, A-20, A-21, A-27 and A-28. Some are inconsistently and arbitrarily identified within the text, e.g., GWe, which is used throughout the draft, is defined in the first paragraph of section 1.2 on page 1.7.
- 1.11 General
The Table of Contents (pp. vii to xi) is too brief for such a large document (over 700 pages plus appendices).
- 1.12 p. 1.1
Clarify the definition of "radioactive wastes" discussed in this document. Traditionally, HLW does not include TRU-intermediate and low-level wastes from the reprocessing plant.
- 1.13 pp. 1.3, 4.22, 4.38
Clarification of the meaning of short-term and long-term as preclosure and post-closure of the repository should be made when the terms are first used. The difference between short-term and near-term is not clear either. On page 1.3, third and fourth paragraphs, the meaning of the "near-term" and "nearer-term" nomenclatures is not clear. On page 4.38, near-term and long-term consequences are mentioned. The explanation for near-term in this paragraph is the same as that given for short-term on page 4.22.
- 1.14 p. 1.3
In presenting the IRG's, "key characteristics of a near-term interim strategic planning base for high-level waste disposal" parts omitted from the "key characteristics" seem significant and should be included.

Comment
Number

It is unnecessary to present the "key characteristics" from both the IRG draft report and the final report (see the FORWARD).

- o The first statement should read (words that are underlined were omitted):

Near term program activities should be predicated on the tentative assumption made for interim planning purposes that the first disposal facilities for HLW will be mined repositories. Several geologic environments possessing a wide variety of emplacement media will be examined Once the NEPA process has been completed, program activities can be tailored accordingly.

- o The footnote on the second statement was deleted. It reads:

The earliest date for operation of a licensed repository, whose site was selected by this process, and using an identical schedule, would be 1992. Actual operation, recognizing reasonable possible deviations from the ideal, could be up to 3 year later.

This says that the actual operation of the repository will be at least 7 years and perhaps as many as 10 years later than the starting date for the first commercial repository (1985) assumed for this study.

- 1.15 p. 1.13

The last sentence in the first paragraph under Site Selection mentions sociopolitical factors as siting constraints to be addressed early in Stage III. What criteria will be used for site screening based on such considerations?

Comment
Number

- 1.16 p. 1.16
 "The conclusion is that the available lethal doses in radioactive waste are far less than the available lethal doses in toxic nonradioactive chemicals now being handled routinely by society as shown in Table 1.3. Further, radioactive wastes decay with time whereas toxic chemicals have no half-lives and hence their quantities remain unchanged with time."
- a. Is the value in Table 1.3 for radioactive waste based on deaths due to the radiotoxicity or the chemical toxicity?
 - b. How does this value behave with time?
 - c. Provide references for Table 1.3.
 - d. Available Lethal Dose is defined as (the number of) potential deaths if dose is uniformly administered."
 - o What does this mean?
 - o What "dose" is uniformly administered?
 - o Administered to what population?
 - e. How many available lethal doses result from the eventual stable daughter products of the radioactive waste.
- 1.17 pp. 1.15, 3.1.62-64
 The section entitled Human Institutions calls attention to the merit of setting up such institutions in the long run control of nuclear wastes. It would be helpful to address whose responsibility it would be to establish and maintain such institutions.
- 1.18 p. 1.17
 The difference between "major disasters" and "primary events" is unclear.
- 1.19 p. 1.19
 Some of the main conclusions given in the summary concerning radiological impacts are not readily traced back to the supporting text, e.g., p. 1.19,

Comment
Number

- lines 14 to 17. The text in the summary section (Section 1.3) states that, "Calculated radiation dose to the total population from routine operations including transportation, assuming that all facilities are located in the same region (a highly conservative and unlikely scenario) amount to no more than about 0.3% of the dose the population would receive from naturally occurring sources and differs by a factor of less than 15 among fuel cycle options." Although the summary gives no reference to where the supporting text for this conclusion is, it appears that the supporting data base is in Tables 3.1.84 to 3.1.87 (summarizing environmental effects from routine operations). However, several entries (e.g., see U and Pu Recycle column on p. 3.1.215) give regional population doses (6×10^4 man-rem) that are greater than 0.3% of background as quoted above.
- 1.20 p. 1.34
Table 1.8. The section entitled Socioeconomic Impact mentions that the impacts were not converted to a 1-5 scale; refers the reader to Table 4.5.2 and states that the impacts are small for all options. It would be helpful to discuss why the 1-5 scale was not used and what rationale was used in both tables to conclude that the impacts would be small. These conclusions appear to be at variance with the statement made on page 1.22 (line 19) which states: "...socioeconomic impacts...could be either small or significant."
- Also refer to the statement on page 3.1.47 (lines 10 and 11) which points out that socioeconomic and political factors may eventually play a determining part in repository site selection.
- 1.21 p. 2.2
 Four impact statements on TRU waste are mentioned as being in preparation by DOE (SRP, INEL, RL, and LASL). Data from DOE received by NRC in conjunction with the DOE licensing study showed TRU waste to exist at ORNL. Will there be an environmental statement for ORNL?

Comment
Number

- .22 p. 3.1.43
In the case of enforcement against private organizations, criminal penalties could be imposed.
- 1.23 p. 3.1.51
The statements that some issues may not be resolved with the necessary degree of certainty seems to conflict with the very next sentence which states that uncertainties can be reduced to acceptable levels.
- 1.24 p. 3.1.60
Line 8 up: The quote of 10 CFR 50, Appendix F, is in error. The regulatory policy stated therein is that liquid wastes at a reprocessing plant must be converted to a dry solid which is "...chemically, thermally, and radiologically stable to the extent that the equilibrium pressure in the sealed container [required before shipping] will not exceed the safe operating pressure for that container during the period from canning through a minimum of 90 days after receipt (transfer of physical custody) at the Federal repository."
- 1.25 Section 3.1.1
This section deals in general terms with geologic considerations. Its deficiencies are mainly in the choice of references used to support the material presented. Seven of the twelve references are contractor reports which have not received proper peer review by the scientific community. One is a working paper by the Interagency Review Group, and two others are elementary geology textbooks. Of these, the one used for data on the chemical composition of rocks was last revised on October 30, 1946, and the other is eleven years old. In most cases, it would have been very easy to use the original references upon which the contractor reports were based. Finally, it would be helpful to cite pages with the references. It would facilitate review of the GEIS.

Comment
Number

- .26 p. A-48
On page A-48, there is an explanatory paragraph on the notations used in the tables. The clarification would not be necessary if powers of 10 were consistently used in all the tables. If this correction is "impossible," the clarification should be made as soon as the first computer print-out, (Table A-14), is used.
- 1.27 Appendix C
Appendix C - The discussion of the "as low as reasonably achievable" principle in this appendix is misleading in that it treats ALARA dose levels as fractions of maximum permissible dose levels for individuals. Instead, ALARA is primarily an analysis of risks to an entire affected population and of the cost-effectiveness of reducing that population risk. While ALARA individual dose limits can be derived for specific activities (e.g., operating nuclear power plants), the most basic ALARA judgment concerns the cost-effectiveness of reductions in overall population risk (e.g., \$1,000 per man-rem in Appendix I).
- 1.28 p. S.1
Specialties of experts that assessed a number of effects are given but it is not stated what the specialties of experts that assessed ecosystem impacts were.
- 1.29 p. S.19
Under Ecosystem Impact it is stated that significant ecological effects may occur from construction of buildings, etc. There is no basis given for this conclusion.

Comment
Number

2. FUEL CYCLE

a. Energy Projections

2.a.1 pp. 1.10, 2.1.2

Considering the growth scenarios on page 2.1.2 and elsewhere (225-400 GWe by the year 2000), would the lower growth scenario change the DOE's approach to repository siting and development? What affect would the lower growth scenario have on the selection of alternatives? Table 1.2 on page 1.10 should also present the repository acreage requirements for a 6300 GWe yr economy.

b. Waste Generation

2.b.1 p. 1.9

Although the number of waste containers shown in Table 1.1 of the GEIS are not unreasonable, some aspects of the table require clarification. First, some of the numbers cannot be derived from Tables 2.1.8, 2.1.10, 2.1.11 and 2.1.13. Secondly, the heading of the third column, or the footnote, should indicate that hulls and hardware are included in TRU intermediate-level waste, if that is the case. Lastly, the last column should indicate that the low-level waste is TRU contaminated.

2.b.2 p. 2.1.4

The GEIS does not address the question of the final disposition of very long-lived fission or activation products, such as ^{129}I , ^{59}Ni , and ^{99}Tc which are separated from TRU or high-level wastes. To help develop national policy for the disposal of these isotopes, cost/benefit estimates of including them with the HLW and TRU wastes should be addressed in the GEIS.

2.b.3 DOE/ET-0028, Section 8

The preliminary information offered by the DOE in Section 8 of the back-up document DOE/ET-0028 is obsolete and does not accurately reflect the

Comment
Number

Pacific Northwest Laboratory studies of decommissioning for the NRC as stated on page 8.1. The NRC information should be properly referenced and the DOE should provide current estimates of the TRU wastes to be expected from all decommissioning activities.

2.b.4 pp. 2.1.3 and 2.1.4

The description of the once-thru fuel cycle is presented. The narrative should address the repository startup and shutdown schedules, i.e., how many will be needed and on what schedule.

2.b.5 pp 2.1.17, 2.1.18, 2.1.19

A review of Tables 2.1.10, 2.1.11, and 2.1.12 shows that the maximum average concentration of LLW at 500 years to be: 2 m Ci/cm^3 for fission and activation products and 0.15 m Ci/cm^3 for actinides and daughters. It is not apparent that it is necessary to send LLW to deep geologic disposal for safe disposal. In view of the large impact the LLW has on repository volume, careful consideration should be given to the need for such disposal and the rationale clearly explained.

2.b.6 p. A.17

Although not implicitly stated, it appears that the inventory in Table A.14 was based on a charge of 3.8×10^5 MTHM. However, the mass associated with the Th-232 (+2 daughters) given in the 1,000,000 year column is 5.8×10^6 MT. There is an obvious error in the program used to generate this table. This single, obvious error brings into question all output generated by the computer program which was used to generate Table A.14.

2.b.7 p. A.58

On page A.58 of the appendix, Table A.52 shows 5760 metric tons of plutonium in spent fuel in the U + Pu recycle mode. Our calculations indicate that this quantity of plutonium indicates an extremely high mix of spent fuel

Comment
Number

from plutonium recycle as compared with UO₂-enriched uranium only fuel. As averaged over the entire time span to the year 2040, the MOX to UO₂ fuel ratio we calculate is 60/40. Please provide your basis for this estimate.

2.b.8

p. A.49

Table A.43, presents the inventories of spent fuel in storage and isolation for the delayed repository availability. By taking the difference between the entries for succeeding years, one should be able to determine the tonnage of spent fuel that is discharged for each year, and from that, the total number of canisters discharged for each year. The following results were obtained:

Year	MTHM	Canisters*
1983	1669	5,619
1985	2979	10,030
1986	1960	6,599
1987	2950	9,932
1988	3430	11,548

*Using 0.297 $\frac{\text{MTHM}}{\text{CAN}}$ per Table 2.1.8

Explain the erratic discharge rates.

Detailed information on the nuclear growth scenario assumed should be provided, including; numbers and types of reactors that come on line each year; and the annual waste streams from the plants, including spent fuel, and low-level waste (volume and activity).

c. Waste Storage and Treatment

2.c.1

p. 1.1

The GEIS should include interim storage facilities in the general description of the fuel cycle since it is apparent from the discussions in the statement that these facilities will be built.

Comment
Number

.c.2

p. 2.1.22

The assumption that spent fuel will be stored after packaging rather than prior to packaging while awaiting shipment to a repository should be justified. Economics may dictate this procedure and it should be based on a cost effectiveness analysis.

2.c.3

pp. 2.1, 2.2

The NRC Final EIS on spent fuel storage, NUREG-0575, should be cited. The NRC Draft GEIS on Uranium Milling, NUREG-0511, April 1979, has been issued for comment.

2.c.4

p. 3.1.184

On page 3.1.184, the following statement is made: "During planned operation of the ESFSF (dry caisson option) no releases of radioactivity would occur." Provide support for this statement.

2.c.5

It appears that all of the below terms refer to the same facility. Terms should be used consistently throughout to avoid confusion and to facilitate comparisons.

packaged spent fuel storage facility (p. 2.1.22)

storage (p. 2.1.22)

offsite storage facilities (p. 2.1.22)

extended storage facility (p. 2.1.25)

storage facility (p. 2.1.25)

(ESFSF) Extended Spent Fuel Storage Facilities (p. 3.1.181)

dry caisson storage facility (p. 2.1.184)

SURF (p. 3.1.184), (p. 3.1.186)

2.c.6

The basis for assuming that two Independent Retrievable Waste Storage Facilities would be needed to serve the needs of the reprocessing industry if repositories are available beginning in the year 2000 should be given. In particular, if economics is the basis, i.e., facility versus transportation costs, such a discussion would assist in any cost/benefit analysis.

Comment
Number

d. Transportation

2.d.1

p. 1.19

The GEIS says transportation risks are the same for all options. However, estimates of risk in the tables in the supporting documents do not seem to support this conclusion.

2.d.2

P. 1.19

It would be useful to provide any available risk (consequence x probability) estimates for the transportation accident being discussed. This will allow a comparison to be made with the risks for the other accident scenarios.

2.d.3

p. D.4

On page D.4, it is stated that the methodology used to calculate the direct radiation dose to persons along the shipping route follows that developed in WASH-1238. Subsequent to the issuance of WASH-1238 an environmental statement on transportation of radioactive material has been published by the NRC. This statement is NUREG-0170, "Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes." The latter document uses a more realistic demographic model in determining population density along the transport route and an improved method for evaluating integrals used in the model. We recommend this refined methodology be used in the GEIS in assessing the radiological impact to the population residing near the transport route.

2.d.4

p. D.5

Population density assumptions used in determining the radiological consequence of transporting radioactive wastes are given on page D.5. Here it is stated that a value of 330 persons per square mile is used for the Eastern U.S. and California and 110 persons per square mile is used for the Western U.S. The environmental aspects presented in DOE/ET-0029, page 4.1.7, use a population density of 90 person per square km (230 persons per square mile). If this is a weighted average, the weighting factors should be given so that their validity can be evaluated.

Comment
Number

2.d.5

Appendix N

Some discussion should be given of the industry's ability to meet the demand for spent fuel casks at the rate they will be required.

2.d.6

Appendix N

The largest accident consequences presented in the GEIS occur during the transportation of radioactive wastes. In the opening paragraph of Appendix N it is stated that much of the detailed analysis is contained in DOE/ET-0029. An examination of these two documents reveals that accident release fractions, curie amounts of isotopes that may be released, and doses to affected individuals are provided. However, some important details concerning accident assumption are not given. These detailed assumptions involve the fraction of released material that is aerosolized and in respirable form. Also missing are resuspension factors. In Appendix B to DOE/ET-0029, reference is given to other reports and computer codes that may contain these factors. These assumptions need to be outlined directly in DOE/ET-0029 so that the degree of realism of the accident analysis can be more easily evaluated and the conclusions compared to other study results.

2.d.7

Appendix N

Throughout Appendix N, the total body radiation dose from the routine transport of radioactive materials is given in various tables. These tables show the dose to the population residing along the transport route and to members of the transport work force. The tables omit the dose to occupants of vehicles using the same route in the case of truck transport. It is not clear whether the dose that results from a delay in transit of the radioactive shipment has been included. These delays could occur from a traffic jam or a stop at a truck stop in the case of truck transport. For rail transport, a delay can be caused by adverse track conditions or a mechanical breakdown.

2.d.8

Appendix N

Some discussion should be included concerning the useful life of spent fuel casks. The analysis appears to assume the casks used in the early

Comment
Number

time frame under consideration, 1980s, will be available for use during the peak years, 2010 to 2020.

2.d.9 Appendix N

Our comparison of the impacts presented in the GEIS with those in DOE/ET-0029, examined spent fuel shipments only. Since it is apparent that in converting results from one document to another several errors have been made, it is recommended that the remaining transportation sections in Appendix N be similarly reviewed.

2.d.10 p. N.1

Table N.1 does not show movement of spent fuel from reactor directly to reprocessing plant which would occur for the recycle options. However, the GEIS states, on page 2.1.5, second paragraph, that it is assumed that storage requirements can be met by power plant storage basins for the recycle options.

2.d.11 p. N.2

It is stated that about 50% of operating reactors do not have rail spurs at the site. The reference system given on page N.3, line 5, shows 90% by rail and 10% by truck. Is this 50% by rail, 40% by intermodal rail and truck, and 10% by truck? Note: On p. N.5, a 45%/45%/10% breakdown is given.

2.d.12 p. N.2

Availability data is out of date. Our most recent information indicates that five NLI-1/2 casks, two TN-8 casks, one TN-9 cask, and two NLI10/24 casks have been built.

2.d.13 p. N.3

line 3: On page 1.11, Section 1.2.2, it is stated that 0.1 rem per year will be used as the background dose rate. Over 70 years this will result in an exposure of 7.0 rem. One percent of this exposure is 0.07 rem. The 0.1 rem the maximum individual receives as a result of transportation is greater than 1% of background exposure, not less than 1% as stated in the GEIS.

Comment
Number2.d.14 p. N.3

The impacts presented in Tables N.3 and N.4 the GEIS are based on a shipping scenario where 100% of all shipments are transported either by rail or by truck. It is not clear whether these impacts are presented only for comparison purposes or whether the scenerios upon which these impacts are based are alternatives to be considered in addition to the reference case. If the latter is true, then the impact of building rail spurs to the 50% of reactors that do not have these spurs should be given in the GEIS. For the reference case, it appears that the impact of transporting the spent fuel by truck from these reactors to the nearest rail siding has not been included in the analysis.

2.d.15 p. N.3, N.4

Impacts presented on page N.3 (Tables N.3 and N.4) and on page N.4 are based on the assumption that all spent fuel is shipped by either rail or truck. Values given in DOE/ET-0029 are based on the reference case of 90% of the spent fuel being shipped by rail from reactors to ISFSFs and 10% by truck with 100% of the shipments from ISFSFs to the final repository being transported by rail. We recommend converting the results presented in Tables N.3 and N.4 and on page N.4 to the reference case so that actual resource commitments can be known and comparison of the GEIS with the back-up documentation can be facilitated.

2.d.16 p. N.3, N.4

It is not clear that the impacts shown in the GEIS have been correctly obtained from DOE/ET-0029. The following discussion develops ratios which can be applied to the results in DOE/ET-0029 to convert them into results that would be obtained if 100% of all shipments are transported by either rail or truck. Following this ratio development discussion is a table outlining some cases where impacts presented in the GEIS appear to have been improperly obtained from DOE/ET-0029.

Table 26.2.3 of DOE/ET-0028 shows 7370 packaged PWR assemblies and 11,340 packaged BWR assemblies needing shipment from ISFSF to a final repository in the year 2000. In the reference case these assemblies would be shipped by rail in a modified NLI 10/24 cask which can accommodate only 7 packaged PWR assemblies or 17 packaged BWR assemblies. (Normally this cask can handle 10 PWR or 24 BWR assemblies.) For truck cask normally only 1 PWR and 2 BWR assemblies can be accommodated. Assuming a modified truck cask can be developed that can accommodate 1 packaged PWR assembly or 1 packaged BWR assembly, the number of truck shipments, for the year 2000, from an ISFSF to a final repository would be $7370 + 11,340 = 18,710$ truck shipments. Table 4.1.1-3 of DOE/ET-0029 indicates a ratio of 120,000 to 1700 for the total number of shipments through the year 2050 compared to the number for the year 2000. Applying this ratio to the 18,710 truck shipments results in a total of about 1.3×10^6 truck shipments through the year 2050 for movement of packaged spent fuel from an ISFSF to a final repository. To determine the total number of truck shipments, the number of truck shipments from reactors to ISFSFs must be added to this value of 1.3×10^6 truck shipments. Table 4.1.2-1 shows 8.9×10^4 truck shipments through the year 2050 for the reference system. Since this reference system is based on only 10% of the reactor shipments being transported by truck, a total of 8.9×10^5 truck shipments would occur if 100% of the shipments were transported by truck. Thus the total number of truck shipments of all types through the year 2050 would be $1.3 \times 10^6 + 8.9 \times 10^5 = 2.2 \times 10^6$ truck shipments. Impacts presented in the GEIS for 100% of all shipments by truck should be $(2.2 \times 10^6) \div (8.9 \times 10^4) \cong 25$ times greater than the impacts given in DOE/ET-0029.

A ratio can also be developed for rail shipments. The reference system has 100% of shipments from ISFSFs to the final repository being transported by rail and no conversion to a 100% rail system is needed here. For shipments from reactors to ISFSFs, the reference system has 90% of all shipments transported by rail. Table 4.1.1-3 of DOE/ET-0029 shows 89,000 reactor shipments, through the year 2050, transported by rail. Since this is 90% of all shipments, an all-rail shipment scenario would have about

99,000 rail shipments. To this total must be added the 120,000 rail shipments from ISFSFs to final repositories, also shown in this table, for a total of 219,000 shipments for an all-rail scenario. For the reference system, the total number of rail shipments is $89,000 + 120,000 = 209,000$ shipments. Thus, the ratio of the number of rail shipments for an all rail shipping scenario to the number of rail shipments in the reference scenario is $219,000 \div 209,000 = 1.05$. Impacts presented in the GEIS for rail shipments should therefore be 1.05 times greater than impacts given in DOE/ET-0029.

A comparison of some of the GEIS results with those presented in DOE/ET-0029 indicates that the ratios developed in the above discussion are apparently the values used in converting impacts from one document to another. For example, on page 4.1.15 of DOE/ET-0029, Table 4.1.2-3 gives a value of 3.1×10^2 man-rem for the dose to the population living along the transport route, through the year 2050, for spent fuel truck shipments. Using the ratio derived in the above discussion, the impact presented in the GEIS, for an all truck shipping scenario, should be 25 times greater giving a value of 7.8×10^3 man-rem and indeed the result given in the GEIS is 8×10^3 man-rem. There are some values, however, that do not agree after this ratio is applied. Cases where there is a lack of agreement between the two documents are outlined in a table which follows the discussion of rail shipments.

For rail shipments, it is more difficult to determine if results for the reference system given in DOE/ET-0029 have been properly converted to an all rail system which is used as the basis for impacts in the GEIS. The difficulty arises because the two systems are so similar and only differ by the 10% of shipments from reactor to ISFSFs that are transported by rail. It is difficult to determine if the ratio of 1.05 derived above has been used or whether a ratio of 1.11 has been used. The 1.11 ratio is obtained from the fact that the reference system has 90% of shipments from reactors to ISFSFs transported by rail, and this may have been improperly applied to the total system to include shipments from ISFSFs to final

Comment Number	Impact	Value	DOE GEIS Location	Value	DOE/ET-0029 Location	Remarks
	Resource commitments, spent fuel rail casks through the year 2050	Values given are identical to those in DOE/ET-0029 90% of shipments by rail	Table N.3 pN.3	Values given are for reference case of 90% of shipments	Table 4.1.1-2, p.4.1.5	Values in GEIS need to be corrected by factor of 1.11
	Resource commitments, spent fuel truck cask through the year 2050 (MT)		Table N.3 pN.3		p.4.1.13	It appears that some ratio different than 25 was used for these resource commitments although no explanation can be found in the document
	Stainless steel	2500 (3000)		120		
	Chromium	450 (550)		22		
	Nickel	200 (240)		9.6		
	Lead	17000 (20000)		800		
	Dose to transport work force through the year 2050, rail shipment, spent fuel (man-rem)	2×10^2 (2.8×10^3)	pN.4	2.5×10^3	Table 4.1.1-7, p4.18	

Comment Number

repositories which for both systems are 100% by rail. In addition, both ratios are close to 1.0 and some results presented in the GEIS have been rounded off, making it difficult to determine which ratio, if any, has been used. For example, the amount of diesel fuel needed through the year 2050 is given in Table 4.1.1-5 of DOE/ET-0029 as $1.7 \times 10^6 \text{ m}^3$. On page N.3 of the GEIS, it is stated that 2×10^6 and m^3 of diesel fuel is needed for an all rail shipping scenario. It is therefore difficult to determine what, if any, ratio was applied to obtain this result. The following table outlines cases where the impacts presented in the GEIS are substantially different than properly converted values obtained from DOE/ET-0029. Values given in parentheses are the results that would be obtained if DOE/ET-0029 values are multiplied by the appropriate conversion factor developed in the above discussion, i.e., 25 for truck shipments, 1.11 for rail shipments. It should be noted that there is one impact where apparently the incorrect ratio of 1.11 was used instead of 1.05. This is the result for nonradioactive effluents released through the year 2050 for spent fuel rail shipments. Table 4.1.1-6 of DOE/ET-0029 shows, for example, that 4.8×10^3 MT of particulates will be released under these circumstances. Applying the incorrect ratio of 1.11 gives a result of 5.3×10^3 MT and this agrees with the result presented in Table N.4 of the GEIS. If the proper ratio of 1.05 had been used, the GEIS result would be 5.0×10^3 MT. This improper ratio has been applied to all the nonradioactive effluents. Since the results are not substantially different and are within the uncertainty of these types of calculations, improper conversions of this type are not included in the following table. It is recommended, however, that for accuracy and consistency, the values given in the GEIS be properly converted.

2.d.17

P. N.4

Repeated reference to discussion of trucker's dose on P. N.4 is misleading. The reference on p. 13 indicates the discussion on p. N.4 explains the overestimate of the dose and the reference on p. N.16 indicates the discussion on p. N.4 is based on experience. The actual discussion on p. N.4 satisfies neither of these descriptions.

Comment Number

- 1.d.18 p. N.4
Lines 5 & 6: Did this result take into account the growth of population along the transport route during the 70 year period?
- 2.d.19 p. N.4
Paragraphs 3 & 4: An inconsistency exists between these two paragraphs. In the third paragraph, it is stated that population doses are calculated based on the permissible limit of radiation. Individual doses given in the fourth paragraph are taken from WASH-1238 which used dose rate values derived from experience rather than permissible limits. In addition, these WASH-1238 numbers were obtained from considering average exposures resulting from the transport of fuel and waste from a power reactor. Since the discussion on page N.4 of the GEIS concerns transportation of spent fuel, it would be better to examine the WASH-1238 analysis of exposures due to transport of spent fuel which can be found on pages 40-42.
- 2.d.20 p. N.4
A reference to NRC/DOT/State surveillance program results would be useful for adding realistic perspective and credibility to the estimates of maximum driver and handler exposure in transportation. See "Summary Report of the State Surveillance Program on the Transportation of Radioactive Materials," NUREG-0393.
- 2.d.21 p. N.4
The transportation accident consequences presented on page N-4 of the GEIS are based on accident number 6.2.8 described in Table 6.2.6 of DOE/ET-0028. Releases of cesium are based on vaporization mechanisms as reported in Supplement II to WASH-1238. A study conducted by Battelle's Pacific Northwest Laboratory, "An Assessment of the Risk of Transporting Spent Nuclear Fuel by Truck," PNL-2588 indicates that other mechanisms can cause additional releases of cesium and other isotopes. These mechanisms involve either oxidation or leaching of the fuel. Releases of radioactive material resulting from these mechanisms can occur in addition to the releases used

Impact	DOE GEIS		DOE/ET-0029		Remarks
	Value	Location	Value	Location	

Dose to transport work force through the year 2050, truck shipment, spent fuel (man-rem)	2×10^5 (3.5×10^5)	PN.4	1.4×10^4	Table 4.1.2-3 p4.1.15	
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Dose to maximum individual from severe truck accident, spent fuel shipment (rem)	24	PN.4	10	Table 4.1.2-9 p4.1.19	
--	----	------	----	--------------------------	--

Accident doses should agree regardless of the percentage of shipment by truck or rail

Nonradioactive effluents, spent fuel rail shipments through the year 2050 (MT)
Alderydes

Organic Acids

No value given for these effluents	Table N.4 pN.3	1.1×10^3 1.4×10^3	Table 4.1.1-6 p 4.1.7
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Comment
Number

in accident number 6.2.8. The probability of accidents occurring where several release mechanisms operate is less than the probability associated with accidents where only a few release mechanisms operate. Thus the risk may be greater for the latter accident than the one involving many release mechanisms. Recommend the GEIS address these accidents that involve several release mechanisms and show that either the risks involved are less than those of accident number 6.2.8 or if the risks are greater, this more severe accident should be used as the umbrella source term for severe accidents.

2.d.22

p. N.4

Although the radiation dose to the maximum individual from postulated accidents are given, the total population dose to persons in the vicinity of the accident is not given. Since this is an important environmental impact, it should be included in the GEIS in context with accident frequencies.

The actual value for this population dose can be found on page 4.1.10 of DOE/ET-0029. The 70 year dose commitment is given as 140 man-rem. Although the analysis uses a population density of 90 persons per square km for routine radiological impacts, the population density used for the accident analysis is 130 person per square km. Note that population densities in suburban or urban areas can be at least an order of magnitude higher than this population density. A severe accident occurring in a suburban or urban area would, therefore, have a substantially greater environmental impact than the accident consequences presented in the GEIS. In order that all relevant impacts be included in the GEIS, recommend including the consequences of severe accidents in high population density areas.

The largest accident dose reported in the GEIS results from a severe accident involving a rail shipment of spent fuel. The resulting whole body dose to the maximum individual is given as 120 rem for a one year period following the accident. The dose is based on the amount of radionuclides released to the atmosphere as given in Table 4.1.1-12 of DOE/ET-0029. The amounts given in this table are based on release fractions

Comment
Number

given in Table 6.2.6 for accident number 6.2.8 in DOE/ET-0028. An examination of the release fractions and cask inventories given in DOE/ET-0028 indicates that the amount of radionuclides given in DOE/ET-0029 and hence the dose reported in Appendix N are in error. There are three sources of error. Mixed fission products and actinides have been excluded from the release, the amount of Kr⁸⁵ released is underestimated, and the amount of Cs¹³⁷ released has been overestimated.

Finally, the following discussion shows that the amount of Cs¹³⁴ and Cs¹³⁷ released for accident number 6.2.8 has been overestimated. The discussion on page 6.2.14 of DOE/ET/0028 indicates that 6×10^{-4} of the cesium inventory may be available for release as a result of fuel rod perforation in a high temperature environment. This result is taken from Supplement II to WASH-1238. According to Table 6.2.6 of DOE/ET-0028, the availability fraction is divided in half between Cs¹³⁴ and Cs¹³⁷. Table 3.3.8 of DOE/ET-0028 shows a cask inventory of 1.7×10^5 curies and 9.4×10^4 curies per MTHM for Cs¹³⁴ and Cs¹³⁷, respectively. Since a cask contains 4 MTHM, this results in 6.8×10^5 curies of Cs¹³⁴ and 3.8×10^5 of Cs¹³⁷ in a cask. Applying the availability fraction of 3×10^{-4} for each isotope yields 204 curies of Cs¹³⁴ and 114 curies of Cs¹³⁷ available for release. Since in accident number 6.2.8, 50% of fuel rods are perforated, this results in 102 curies of Cs¹³⁴ and 57 curies of Cs¹³⁷ being released in this accident. Table 4.1.1-12 of DOE/ET-0029 shows 200 curies of Cs¹³⁴ and 110 curies of Cs¹³⁷ being released. Perhaps the fact that only 50% of the rods are perforated was not taken into account.

We recommend that the radiation dose to the maximum individual resulting from this accident be reevaluated in light of the above comments.

2.d.23

p. N.4

The consequences presented in page N.4 for severe accidents are based on the dose received by persons from radionuclides released to the atmosphere. Since severe accidents may cause a reduction in shielding efficiency, doses resulting from radiation emanating directly from the cask should

Comment
Number

also be evaluated. The description of severe accidents in Table 6.2.6 of DOE/ET-0028 indicates a small opening will exist in the cask.

Accident number 6.2.8 is based on number 6.2.7. Number 6.2.8 assumes that no emergency action is taken to cool the cask involved in the 6.2.7 accident. This results in 50% of the fuel rods being perforated in number 6.2.8 instead of only 1% being perforated in number 6.2.7, in addition to 100% of the coolant being released in both accidents. Thus, release fractions in number 6.2.8 should be 50 times higher than in number 6.2.7. Indeed, for Kr⁸⁵, I¹²⁹, and H³ the release fractions given are 50 times higher for number 6.2.8 than for 6.2.7. However, although mixed fission products and actinides are reportedly released in number 6.2.7, only Cs¹³⁴ and Cs¹³⁷ are reported as being released in number 6.2.8. This can also be seen in Tables 4.1.1-10 and 4.1.1-12 of DOE/ET-0029 which gives the actual number of curies released. Note that Table 4.1.1-10 gives the curies released for accident number 6.2.6, a moderate accident in which only 5% of the cavity coolant is released and only 0.25% of the fuel rods exhibit cladding failure. The table shows fission products such as Sr⁹⁰ and Nb⁹⁵ and the actinides such as Pu²³⁹ and Cm²⁴² being released. Table 4.1.1-12, which lists the radionuclides released for accident number 6.2.8, the severe accident, does not contain any of the additional fission products or actinides listed for the less severe accident. Is this simply an oversight or is the contribution to the dose from these nuclides negligible compared to the dose resulting from the nuclides that are listed?

A study conducted by Battelle's Pacific Northwest Laboratory, "An Assessment of the Risk of Transporting Spent Nuclear Fuel by Truck," PNL-2588 uses release fractions for actinides and fission products other than gases that are significantly higher than those derived from the accidents described in DOE/ET-0028. As previously shown the release fractions for actinides and mixed fission products in accident number 6.2.8 should be 50 times higher than those used in accident number 6.2.7. Table 6.2.6 of DOE/ET-0028 shows a release fraction of 1×10^{-8} for actinides and mixed fission products for accident number 6.2.7. The release fraction for accident number 6.2.8

Comment
Number

should therefore be 5×10^{-7} . Table 8.2 of PNL-2588 shows a release fraction of 2×10^{-5} for actinides and other fission products. Note that both of these release fractions are for accident scenarios that involve creep rupture of fuel rod cladding. Since in accident number 6.2.8 it is assumed that 50% of the rods fail the release fraction for actinides and other fission products should be 1×10^{-5} if the results of PNL-2588 are used. Recommend the basis for the release fractions used in DOE/ET-0028 be reexamined and any discrepancies with the fractions used in the PNL study be resolved.

The following discussion shows that the amount of Kr⁸⁵ released for accident number 6.2.8, the most severe accident, has been underestimated. Table 6.2.7 of DOE/ET-0028 indicates that 30% of the Kr⁸⁵ will exist in fuel rod void spaces. Accident number 6.2.8 assumes that 50% of the fuel rods are perforated so that the release fraction reported in Table 6.2.6 of DOE/ET-0028 is 0.15. This table also indicates that the cask inventory given in Table 3.3.8 of DOE/ET-0028 should be used for determining the actual number of curies released. Table 3.3.8 indicate 9.5×10^3 curies per MTHM. Since Table 6.2.6 indicates that a cask will contain 4 MTHM, this means a total inventory of 38×10^3 curies of Kr⁸⁵. With a release fraction of 0.15, this results in 5.7×10^3 curies of Kr⁸⁵ being released. Table 4.1.1-12 of DOE/ET-0029 shows only 5.3×10^3 curies of Kr⁸⁵ being released.

2.d.24

p. N.5

Paragraph 3: Some discussion should be included describing the composition of a special train and the advantages and disadvantages resulting from its use. Is it a safer mode of transport? Does it have better safeguard features?

2.d.25

p. N.6

Line 23: Can an aerial radiation survey detect a spent fuel cask that has not been breached and is located inside a building?

Comment
Number

- 2.d.26 p. N.7
The statement on prime considerations may be misconstrued. The prime safety considerations in transportation packaging are containment, shielding, and subcriticality. Heat dissipation is not a prime safety consideration but is important to the performance of the other safety features.
- 2.d.27 p. N.7
Why is the cask maximum thermal design load set at 50 kW?
- 2.d.28 p. N.9
Last paragraph: The accident postulated here results in 37 rem to the total body. Table 3.1.88, page 3.1.224, shows the results of the worst design basis accident, which for SHLW - severe impact and fire is 7 rem.
- 2.d.29 p. N.13, 16, 21
On pages N.13, N.16, and N.21 reference is made to page N.4 and a discussion on dose to truckers. The reference on page N.13 indicates the discussion on page N.4 explains the overestimate of the dose and the reference on page N.16 indicates the doses discussed on page N.4 are based on experience. These references are misleading since the discussion on page N.4 satisfies neither of these descriptions. Is the reference intended to apply to WASH-1238 which is the basis for the truckers dose given on page N.4?
- e. Safeguards
- 2.e.1 General
The uranium-only fuel cycle is not addressed from a safeguards standpoint although the health, safety and environmental aspects of the U-only fuel cycle are discussed. In addition, although the basic purpose of a safeguards system is identified, there is no discussion of the concepts or elements of safeguards systems potentially applicable to each waste form and storage mode. Also, the draft GEIS does not identify how much effort would be needed to mine the waste nor does it address the issue of how the repository management would assure the public that all waste material is in its

Comment
Number

- authorized location if faced with a blackmail threat after closure of a repository. Finally, the draft GEIS should make clear the kind of adversary that is considered when a safeguards system is designed.
- 2.e.3 p. N.6
The NRC has promulgated an interim rule on physical protection of spent fuel shipments (Federal Register 44, 34466 (June 15, 1979). Accordingly, the footnote is no longer valid.
- f. Other Fuel Cycle Alternatives
- 2.f.1 p. 1.10
Allusion to Alternate Fuel Cycle
On page 1.10 (and elsewhere) the statement is made that a separate and distinct nuclear fuel cycle might be in existence to receive 1300 metric tons of plutonium by the year 2040. This "Alternative" fuel cycle would also produce radioactive waste. Although this disposition may appear to be possible, the more prudent approach would be to consider this excess plutonium to be TRU waste requiring safe disposal in a repository. However, if credit is to be taken for use of the plutonium in this "alternative" fuel cycle, the disposal of radioactive wastes from this fuel cycle should be discussed.
- 2.f.2 p. 3.1.226
The discussion on page 3.1.226 of other fuel cycle alternatives is out of place in the section on geologic disposal impacts. There is no relationship drawn between the impacts shown in this section and this discussion of other fuel cycles.

Comment
Number

3. CONVENTIONAL GEOLOGIC DISPOSAL

3.a Siting

3.a.1 p. 1.5

The discussion of multiple barriers should also indicate the barrier-like effect of the liquid transport processes that result in dilution and dispersal of radioactive material. While these processes are technically not barriers, they serve almost the same function by reducing the amount of material reaching a specified point and by increasing the time for a specific quantity of material to reach a location.

3.a.2 p. 3.1.5

The section on Hydrology of Host Rock is highly simplified. For more detailed explanations standard reference works such as the following should be cited:

Bear, J., Dynamics of Fluids in Porous Media, American Elsevier, New York, 1972

Bear, J., Hydraulics of Groundwater, McGraw-Hill, New York, 1979.

Bouwer, H., Groundwater Hydrology, McGraw-Hill Book Co., New York, 1978

Davis, Stanley N. and R. J. M. DeWiest, Hydrogeology, J. Wiley and Sons, New York, 1966.

DeWiest, Roger J. M., Flow Through Porous Media, Academic Press, New York, 1969.

Domenico, Patrick A., Concepts and Models in Groundwater Hydrology, McGraw-Hill, New York, 1972.

Freeze, R. Allan and John Cherry, Groundwater, Prentice-Hall, 1979.

International Association for Hydraulic Research, Fundamentals of Transport.

Phenomena in Porous Media, Elsevier Publishing Co., New York, 1972.

Johnson Division, Universal Oil Products Co., Ground Water and Wells, St. Paul, Minnesota, First Edition, 1966, 2nd Printing, 1972.

Comment
Number

Lohman, S. W., Ground Water Hydraulics, U.S.G.S. Professional Paper 708, U.S. Government Printing Office, Washington, D.C., 1972.

Polubarinova-Kochina, P. Ya., Theory of Groundwater Movement, Princeton University Press, Princeton, New Jersey, 1962.

Walton, W. C., Groundwater Resource Evaluation, McGraw-Hill, New York, 1970.

3.a.3 p. 3.1.10

The reference cited (#8) for Figure 3.1.1 is incorrect for this figure. The information is not found in that report. However, there is an identical map in Y/OWI/TM 36/3. This was derived from USGS Bulletin 1148. The original reference should be used especially since it is readily available to the public whereas the contractor report is not.

3.a.4 p. 3.1.11

An explanation should be provided as to why the areas shown in Fig. 3.1.2 are favorable granitic sites. Certainly there are many other areas of the U.S. where granitic rocks are either at, or close to the surface, e.g., St. Francois Mts. of Missouri, Llano uplift of Texas, Wasatch and Uinta Mts. of Utah, the Big Horns of Wyoming, and the shallowly buried part of the Nemaha Ridge.

3.a.5 p. 3.1.11

The reference cited for Fig. 3.1.2 is incorrect and it could not have been developed from the information found in Reference 9. However, it appears in Y/OWI/TM 36/3 and is based on a diagram in OWI-76-27. Original sources should be used.

3.a.6 p. 3.1.12

p. 7.2.9, DOE/ET-0028

It would be better to use either a more recent reference than Pirsson's 1947 book or to be more selective in the data excerpted from Pirsson. For example, the silica content of the granite is rather high. It turns out

Comment
Number

that this represents a single sample from Pikes Peak (Pirsson, pg. 169) It would have been better to use Tschirwinsky's average of 90 analyses (Pirsson, pg. 169) which results in a significantly different chemical composition for an "average" granite. An alternate source of information is Clark, S.P., 1966, "Handbook of Physical Constants," Geol. Soc. of Amer.

3.a.7 p. 3.1.13p. 7.2.10, DOE/ET-0028

Figure 3.1.3 does not appear in Reference 10 as indicated in GEIS. Apparently it was adopted from a similar but slightly different diagram in Y/OWI/TM 36/3 which was adapted from OWI-76-26 and Tourtelot 1962. The original references should have been used. The figure's caption implies that it shows all the shale formations (sic) in the U.S.. However, there are a number of significant omissions such as the thick shales found in the Appalachian foldbelt, the Ouachita Mts., Anadarko Basin, and the Midland, Marfa, and Delaware Basins of West Texas.

3.a.8 p. 3.1.14

The table is very important since it is a direct numerical comparison of major host rock types. Therefore, the selection of numerical ratings for each characteristic of each rock type should be discussed.

3.a.9 p. 3.1.14p. 7.2.13, DOE/ET-0028

Figures 3.1.4 and 7.2.4 are incorrectly referenced, are incorrect and misleading:

1. They fail to show some of the other basalt areas which should be assessed as candidates for deep geological burial of HLW, e.g., Colorado Plateau, Rio Grande Valley, San Juan Mts. of Colorado, Snake River Plains, Triassic Basins of the Carolinas, Virginia and Pennsylvania.

Comment
Number

2. The Keweenaw Series is misplotted as is the Triassic of N.J. and Connecticut. This is not surprising as the map of the Keweenaw which was supposedly used in compiling this map (Y/OWI/TM 36/7, Figure 3-1) is illegible.

3. Reference Y/OWI/TM 36/7 is cited as a source of information for the location of the Triassic "Lavas." There is no information on the Triassic in this publication.

4. The expression Keweenaw and Triassic "Lavas" is misleading, as many of these basalts are not extrusive igneous rocks, e.g., Palisades Sill.

5. Figure 7.2.4 could not have been developed from information found in Y/OWI/TM-44.

3.a.10 p. 3.1.32

The first paragraph does not reflect fissure and joint permeability differences, and induced characteristics due to construction. The final paragraph makes an assertion that is not supported; i.e., no bases have been provided to support the conclusion that groundwater inflow can always "...be controlled by state-of-the-art engineering technology."

3.a.11 p. 3.1.67

The age of the earth is given as 10 billion years. The current geologic estimate of the age is 5 billion years.

3.a.12 p. 7.2.2, DOE/ET-0028

The statement that: "The repository should not be sited in or near an area in which igneous or volcanic activity has occurred during the post-Pleistocene" should be assessed and actively discussed by DOE. An assessment should be made of the potential for volcanic activity and its impact

Comment
Number

on repository performance. The assessment should estimate the actual effects, detrimental or beneficial, on repository performance by different types of eruptions.

3.a.13 P. 7.2.3, DOE/ET-0028

The credibility of section 7.2.2 is weakened by either a lack of documentation for the statements (e.g., see pg. 7.2.6 Southwest Florida) or the use of very old references (e.g., see 7.2.6 para. 3 on the Supai Formation of the Hölbrook Basin of Arizona) when more recent material should be available.

3.a.14 p. 7.2.3, DOE/ET-0028

Figure 7.2.1. was adapted from Y/DWI/TM-44, which was adapted from Pierce and Rich, USGS Bulletin 1148. The original source should have been used in developing this figure.

3.a.15 p. 7.2.3, DOE/ET-0028

The geologic term "Formation" is misused throughout the GEIS. Although this appears to be a minor editorial comment, it may have legal ramifications. The term is defined in the American Code of Stratigraphic Nomenclature which is to be found in the Bulletin of the American Association of Petroleum Geologists (1961, pp. 645-660). Basically, a formation is a specific rock unit which has distinctive lithologic characteristics which allows it to be mapped. Sandstone, limestone, shale, granite and basalt are not formations; whereas rock bodies such as the Dakota Sandstone, Salem Limestone, Pierre Shale, and Louann Salt are.

3.a.16 p. 7.2.8/7.2.9, DOE/ET-0028

The statement that the "mineral components of granite are almost inactive chemically under ambient temperature and pressure conditions" is misleading. Granite does decompose at surface temperatures and pressure as evidenced by well developed regoliths found on top of many granites.

Comment
Number

3.a.17 p. 7.2.9, DOE/ET-0028

The statement that igneous rock "...range in chemical and mineralogical composition from granite to closely related rocks such as granodiorite" is technically true but misleading. The range goes far beyond granodiorite through gabbro to pyroxenite and dunite.

3.a.18 p. 7.2.9, DOE/ET-0028

The statement that granite has "...little ability to deform under stress..." is not true. Under varying combinations of the following: (1) high confining pressure, (2) elevated temperatures, or (3) when the stresses are applied for long time spans, granite will deform.

3.a.19 p. 7.2.9, DOE/ET-0028

The statement that "granite is mostly composed of silica and mica" is misleading. Mica makes up a small percent of most granites and quartz rarely exceeds 30%. Mention should be made of other minerals common in granite such as the feldspars and ferromagnesian minerals.

3.a.20 p. 7.2.10, DOE/ET-0028

The basic references of Pirsson 1947 and Gilluly, Woodford, and Waters, 1968 should be replaced by reference to one of the following: Robert L. Folk's Petrology of Sedimentary Rocks (Hemphill's, Austin, Texas), Blatt, Middleton, and Murray's Origin of Sedimentary Rocks Prentice-Hall or Pettijohn's Sedimentary Rocks, Harper Brothers, N. Y.

3.a.21 p. 7.2.10, DOE/ET-0028

Contrary to line 5, Table 7.2.1 gives no direct information on the mineral content of shales.

3.a.22 p. 7.2.12, DOE/ET-0028

The statement that basalt is an "extrusive volcanic mafic (high in magnesium rock silicates) rock" is doubly misleading: (1) Not all basalts are extrusive, e.g., Palisades Sill, and (2) the mafic minerals are not limited to magnesium silicates.

Comment
Number

- 3.a.23 p. 7.2.12, DOE/ET-0028
Section 7.2.2.4 leaves the impression that basalt has a low permeability. There are many situations in which this is not the case, e.g., Idaho Falls, or the domestic water sources on the Hawaiian and Canary Islands.
- 3.b Waste Form and Packaging
- 3.b.1 p. 4.1.30
The 3rd paragraph states that fluid bed calcination was identified in a previous report (ERDA 76-43, Vol. 2, Chapter 6) as being the most well-developed calcination process and, therefore, has been selected as the reference calcination process for this report. However, page 4.1.4 states that the reference vitrification process is spray calcination/in-can melting. The report should clarify why one calcination process is referenced to make glass waste form and another to make a calcine waste form.
- 3.c Design and Operation
- 3.c.1 p. 1.9/1.10
The staff has attempted to corroborate the numerical values given in Tables 1.1 and 1.2. In attempting to understand the bases for the tables and the connection between them, other parts of the GEIS and the supporting documents were searched. Neither the numerical values nor the relationship between the two tables could be substantiated. Therefore, we recommended that you provide a detailed explanation in the GEIS text of the method whereby the numerical values of Tables 1.1 and 1.2 were developed. It would be helpful if intermediate tables were prepared which indicate how Table 1.1 relates to Table 1.2. The headings (or footnotes) on Table 1.1 should clearly indicate what HLW or TRU wastes are included in each column.
- 3.c.2 p. 2.1.22
The rates of receipt of spent fuel at the repository are presented. The maximum receipt rate of 12,000 MTHM/yr converts to 40,400 canisters/yr (assuming 0.297 MTHM/canister per Table 2.1.8).

Comment
Number

- According to Section 7.4.5 of DOE/ET-0028, the repository is operating around the clock for a total of 175 days/year or 4200 hours/yr. This means that canisters are disposed of at a rate of 10 per hour, when 12,000 MTHM/hr are received. Considering the remote handling required these rates appear unrealistically high. The design of handling systems to accomplish this should be presented.
- 3.c.3 p. 3.1.30 and 3.1.116
GEIS states that: "The effects of rock discontinuities on rock strength are difficult to evaluate..." "A structural system of grouted rock bolts, wire mesh and shotcrete effectively support this type of ground (shale)." "The shale surfaces can be protected...to prevent slaking" (p. 3.1.30).

Ground support in a shale repository at depths of about 600 m is likely to be a major and costly problem. It is more likely that reinforced concrete shields will have to be used extensively in all main corridors, crusher rooms and places that have to be kept open for retrievability. Support system will likely be similar to that used in the Washington Metro or in the Clear Creek Tunnel. The support costs on p. 3.1.116 for shale are clearly underestimated.
- 3.c.4 p. 3.1.31
"Under high stresses and temperatures the room closure rates may be high... engineered support would be necessary... A support system can be provided..."

Since closure rates are expected to be high, the GEIS should describe the support systems and expected closure rates and the effectiveness of the support systems.
- 3.c.5 pp. 3.1.35, 3.3.12, K.2, K.3, K.11
pp. 7.3.6, 7.3.12, 7.4.2; DOE/ET-0028
The GEIS evaluates the impacts of 5 year and 25 year retrievability. Given the uncertainties concerning geologic emplacement in mined repositories raised by the IRG and the GEIS which must be addressed by site specific

Comment
Number

in-situ tests, it appears prudent to provide capability for retrievability of the wastes for the normal operating life of the repository and for as many years thereafter as may be needed to retrieve the emplaced wastes. The GEIS should address retrievability in a fashion that the potential for such retrievability can be properly assessed.

The following examples are cited:

A conclusion that readily retrievable conditions in a repository in salt would prevail in storage rooms for at least five years is based on linear thermomechanical analyses. The time behavior of salt under thermal and mechanical loading cannot be approximated as a linear relationship. The GEIS makes an allowance of two feet to accommodate for expected closures (expected closures are not specified).

Project Salt Vault, Chapter 12 presents several figures (Figure 12.36, etc.) that describe pillar behavior as a function of mechanical and thermal loading and time. A 50% shortening of a pillar is expected when subjected to a load due to 8000 psi stress and 100°C temperature during a period of 200 hours. Similar results presented are 28% shortening under 6000 psi at 22.5°C for 30,000 hours and 45% shortening under 200 psi at 200°C for 500 hours. The behavior exhibited is not linear and a significant under-estimation of closure will be obtained if a linear approximation is used.

If later studies have been used to discount the results of Project Salt Vault, they have not been identified in GEIS. Consideration of the creep behavior of salt under thermomechanical loading is important for the design of a repository in salt because it will affect the short (operational) and long (retrievability) term stability of storage rooms and access ways. If stability is compromised, the integrity of the impermeable barrier between the salt bed and overlying aquifers that is assumed by GEIS, may be compromised. This would lead to problems associated with groundwater movement in salt that have not been addressed.

Comment
Number

Based on Project Salt Vault data (ORNL-4555, Chapter 12) the mine layout design postulated by GEIS for a repository in salt would therefore be inadequate. This would necessitate a different design that would consider the thermomechanical effects on salt mass behavior. Extraction ratios may have to be reduced, room and pillar dimensions changed, etc.

3.c.6 pp. 3.1.28, 3.1.36, 3.1.41, 3.1.120, 8.4.7

The subject of occupational radiation exposure is not adequately addressed in the GEIS. It should be considered in connection with short term environmental impacts and the probability of various accidents occurring during the handling and emplacement of waste canisters.

3.c.7 p. 3.1.36

The statement is made in the fourth paragraph that the costs for additional support necessitated by the reduction in rock strength due to radiation, are not expected to be significant. The basis for this conclusion should be presented.

3.c.8 p. 3.1.37

The statement, "... maintaining retrievability longer than needed to reasonably assure repository operation increases the occupational and general populace risk." is unsubstantiated.

3.c.9 pp. 3.1.104, 3.1.107, 3.1.110, 3.1.112, K.17, K.113, K.115
pp. 7.2.22, 7.4.4; DOE/ET-0028

A major deficiency in the design of the repositories in granite, shale, and basalt is that they have been designed as if the host rock were salt. The repositories in the four geologic media should not be of similar design. For instance, the inherent structural characteristics of granite have not been taken into consideration. The design of a mine in hard rock is substantially different from that in salt. Where, by the nature of the material, a repository in salt is confined to a single level, a repository in massive granite need not be. The long term stability of large rooms in granite is well known. Transportation could be by track systems - either

Comment
Number

Comment
Number

conventional or suspended on roof mounted tracks; manually operated or remote controlled. Rooms could be in the traditional orthogonal pattern as presented or they could take on different configurations. The alternative repository layout possibilities in granite are not addressed in TM-36. In general, the preconceptual repository design procedure is not clear, and lacks a logical, consistent argument. Little attempt is made to evaluate the design either in part or in total, in terms of the risks associated with nuclear waste storage, particularly the long-term containment aspects. Therefore, it is difficult to judge the adequacy of proposed design measures and safety features.

3.c.10

p. 3.1.116

It is difficult to comment on the resource requirements presented in Table 3.1.11 without having access to their back-up. It is, however, strange that construction steel, lumber and concrete costs per MTHM for granite are greater than those for salt. Granite is structurally far superior to salt, has no creep characteristics and relatively lower risk of being inundated by water from an overlying aquifer. Retrievability in granite should be easier.

Support requirements in salt, in order to maintain access to the storage rooms during the retrievable period is expected to be considerably greater than in granite and basalt. The problem in salt is compounded by a high level of uncertainty regarding the behavior of the salt rock mass when subjected to thermal and mechanical loading.

It appears that the resource requirements are biased in favor of salt due to poor design of repositories in other media. The differences between the unit resource figures for salt and those for granite and basalt are not adequately justified.

3.c.11

p. 3.1.133

The statement that "granite unit costs are less than those for shale" is inconsistent with the data presented in Table 3.1.28 on page 3.1.134.

.c.12

p. K.5, Appendix K

The design of the repository used in the GEIS is a single level room and pillar mine for all media and waste types. Thermal criteria are then used to set capacities for each medium and fuel cycle. Optimization of the design for a given waste type in a particular medium would likely result in different capacity estimates.

3.c.13

P. K.8

Figure K.6 shows a smaller temperature increase after emplacement of waste for a repository in shale (Figure K.6, page K.8) than for a repository in salt (Figure K.2). The opposite should be true because the temperature increase should be inversely proportional to the thermal conductivity and shale has a lower thermal conductivity than salt as shown in Tables 7.2.6 and 7.2.3, respectively.

3.c.14

p. K.19, Appendix K

It is stated that 25-year retrievability requires lower thermal densities. For salt and shale the decrease is a factor of 2 while for granite and basalt it is 2.5. Hence costs increase by the same factor. The reason given is the need to maintain room and pillar stability for 25 years. Why is the effect greater for granite and basalt?

3.c.15

p. 7.1.2 and 7.2.18, DOE/ET-0028

"...there were no immediate detrimental effects on the stability of salt as a result of exposure to heat or radiation"

"The physical behavior of salt is drastically affected by temperature. ...for a rise in temperature from 20°C to 100°C the strain increased by a factor of seven."

The GEIS should discuss whether retrievability in salt can be guaranteed under the expected thermal loadings. It should also discuss whether the integrity of seals in the salt repository can be maintained following closure.

Comment
Number

.c.16 pp. 7.2.4, 7.3.16; DOE/ET-0028

"(Creep) is difficult to stabilize in tunnel openings."

"From experiments....equations can be developed to describe the creep behavior of salt"

Since equations have been developed which describe the behaviour of salt a posteriori, the GEIS should discuss whether or not they can predict the behaviour of salt under thermomechanical loading conditions?

3.c.17 pp. 7.4.24/25; DOE/ET-0028

Several potential occupational and environmental hazards are associated with the ventilation design as described in DOE/ET-0028 and in Y/OWI/TM-36. The following questions need to be considered:

- o What are the risks of escape of radionuclides via the fresh airway as a consequence of a transportation accident underground?
- o How will the integrity of seals between fresh airways and storage rooms be maintained if closures of up to 2 feet are expected in a repository in salt?
- o What are the risks associated with backfilling and retrieval operations?
- o What measures will be taken to reduce respirable dust to acceptable concentrations? Note that mining in granite will have an associated health hazard due to the siliceous type of dust generated. What are the expected health effects due to dust?

3.c.18 p. 7.4.30, Table 7.4.11, DOE/ET-0028

No estimate of occupational risk is included in the accident analyses nor is there any discussion of possible impacts on continued repository operation, repository closure or retrieval of waste already emplaced. For example what will such impacts be for accident 7.5 or 7.6.

Comment
Number

For Accident 7.6, the safety system is a failsafe wedge type braking system on the cage. What is the maximum allowable braking distance of the cage for the expected release?

3.c.19 Appendix 7A, DOE/ET-0028

The tables in this Appendix present the mining and construction costs for a repository and its support facilities (surface). Operating and decommissioning costs for a repository should also be given and taken into account in comparing the alternatives. The GEIS should consider all costs that will be incurred through repository closure.

3.c.20 Y/OWI/TM-36

TM-36 lacks supporting analyses for salt. For example:

Hydrology: Volume 21 "Ground Water Movement and Nuclide Transport" addresses granite, shale and basalt - no salt.

Thermomechanical: Volume 20 "Thermomechanical Stress Analysis and Development of Thermal Loading Guidelines" addresses granite, shale and basalt - no salt.

3.c.21 General Comment

The problems associated with retrievability have not been adequately discussed in GEIS. The following are specific areas of concern regarding retrievability.

- o Occupational hazards associated with retrieval options. Depending on the time delay between retrieval and emplacement operations and the geologic medium of the repository, some canisters may be corroded, damaged or stuck (due to deformation or spalling host medium) such that there will be a risk of exposure to the workers involved in the retrieval operations. There could also be a risk of escape of radionuclides into the biosphere if the integrity of seals separating main airways from storage rooms have not been maintained.

Comment
Number

If overcoring is necessary to remove canisters, activation of the disposal media may result in radioactive dust. Occupational exposures should be estimated.

- o In order to have retrievability, all main entries (corridors), storage rooms and exhaust airways need to be kept open. Based on present day mining technology, this should not be a problem in granite and basalt. However a repository in shale will require massive support requirements to maintain retrievability and retrievability in salt is questionable. There is significant evidence that salt rock behavior under thermal and mechanical stress is such that rapid closure rates can be expected. It may be impossible to maintain integrity of seals under such closure rates. (Closures of 2 feet may reasonably be expected - TM-44, Table 5.12).

3.c.22 General Comment

The rationale for the thermal and thermomechanical limits on which repository designs are based is missing from GEIS and should be provided.

3.c.23 General Comment

The statement is made that criteria for the performance of the mined repository have not yet been established. Instead several local criteria such as limits on thermal loading, limits on area of the repository, limits on geometry (single level repository) etc. have been imposed on the design process. This appears to be a process of local optimization. It appears that imposing these limits on different geologic media results in noncomparable containment of the waste. For example: With the design process and argument presented in GEIS, would a repository in granite 200 m below the surface contain the wastes with the same level of effectiveness as the repository in salt at a depth of 580 m? Given the knowledge that large chambers in granite are feasible, different repository designs and waste storage designs should be considered.

Comment
Number

3.c.24 General Comment

The discussion of costs and capacities for each medium and fuel cycle is confusing. Some of the data appears contradictory. For example:

Table 1.5 (GEIS)

Unit power costs (5 year retrievability) mill/kwh

	Spent Fuel	U + Pu recycle
Salt	0.45	0.50
Granite	0.51	0.58
Shale	0.46	0.59
Basalt	0.53	0.63

Table 3.1.26 (GEIS)

Construction Costs Including Decommissioning 10⁶ \$ (1978)

	Spent Fuel	U + Pu recycle
Salt	1000	1200
Granite	2600	2000
Shale	1300	1300
Basalt	3100	2300

Table 1.2 (GEIS)

Total Repository Acreage Required for 10,000 GWe-y Economy

	Spent Fuel	U + Pu recycle
Salt	16,000	12,000
Granite	6,000	12,000
Shale	12,000	20,000
Basalt	6,000	12,000

Comment
Number

Comment #1 - The unit power costs do not appear to reflect the construction costs. Consider the U + Pu recycle case in which, according to Table 1.2, salt, granite and basalt each require 12,000 acres. The construction costs for granite and basalt are almost double that of salt but the unit costs reflect only 20 percent changes.

Comment #2 - The difficulties of mining in granite and basalt are comparable. Why does a repository in basalt cost \$500 million more than in granite?

Comment #3 - If we compare construction costs and note that the only apparent difference is that fewer holes will be required in the U and Pu recycle case, then Table 3.1.26 is puzzling. Why, for example, does salt cost \$200 million more, granite cost \$600 million less, basalt costs \$800 million less and shale has no difference?

3.d Safeguards

d.1 p. 1.23

First sentence, second paragraph, and the last sentence, third paragraph are assertions that are not backed up by analyses in this section or in later sections. They should be substantiated.

3.d.2 p. 1.23

Second sentence, second paragraph. From a sabotage standpoint, high-level waste without plutonium might also be an attractive material and should be included in the list of material in this sentence.

3.d.3 p. 4.16

Section 4.5.4, Safeguards and Security is incomplete for several reasons. The GEIS has been prepared for decision makers and the public. The uranium-only recycle has not been addressed in this draft GEIS from a safeguards standpoint. This and other cycles could have significant safeguards implications. In addition, this section attempts to identify the purpose of proposed safeguards systems but does not provide the decision maker or

Comment
Number

public with a discussion of the concept or elements of proposed systems for specific forms of waste. It will be difficult to form a judgment on the adequacy of any safeguards system without this information.

3.d.4 p. 4.16

The footnote at the bottom of page 4.16 is not accurate. The Nuclear Regulatory Commission is studying this problem but has not yet published safeguards requirements specifically applicable to waste repositories.

3.d.5 p. 5.7, Appendix S

The uranium-only cycle should be included in the discussion and factors of attractiveness should be identified for this cycle. Because of the presence of plutonium in this cycle the sabotage and the theft susceptibilities should be analyzed separately.

Consequences and environmental impacts of successful acts of dispersal, sabotage or theft have not been considered in establishing the susceptibility index. These factors could have a bearing on the level of safeguards required in factor number 3 in the short-term susceptibility to encroachment case.

The level of safeguards appropriate for a type of waste appear to be based upon an evaluation concerning the types of wastes which would be attractive for theft or sabotage. This attractiveness criterion is inherently conjectural and should not be used as a basis for determining safeguards requirements. The appropriate considerations in this area are the potential consequences to public health and safety and common defense and security that result from successful theft or sabotage of each specific type of waste.

3.d.6 p. 5-11, Appendix S

Table S.3 "Proposed Safeguards Requirements" does not include any material control and accounting requirements. Safeguards requirements for a high-level waste repository might include some form of accountability requirements

Comment
Number

during the period prior to final closure, particularly in the case of the uranium-only fuel cycle where significant quantities of plutonium would be present.

3.d.7

Appendix S

The Safeguards and Security section of Appendix S is incomplete for several reasons. The section does not address safeguards requirements for the uranium-only cycle although the GEIS includes discussions of this cycle in other areas of the statement. In addition, although a safeguards group evaluated and ranked various waste management systems from a safeguards susceptibility standpoint, there is no discussion of the methodology used by the group to arrive at the group conclusion. Thus, the work of the group cannot be evaluated.

3.e Short-Term Environmental Impacts

3.e.1

p. 3.1.41

The listed impacts are essentially written off without any perceived bases. For example, storage and disposal of mined mineral on the surface is a visual as well as potential biological impact. These impacts should be fully considered and analyzed in a generic manner, and not be left for a later determination.

3.e.2

p. 3.1.115

The surface storage of mined material is not sufficiently evaluated as an environmental impact. A more detailed impact analysis of surface storage should be provided and cross referenced whenever it is discussed.

3.e.3

p. 3.1.115-3.1.136

No discussion of the hydrologic design criteria of the surface facilities is given. If the site is to be designed to withstand the Probable Maximum Flood, so state and discuss. If not, discuss the consequences of a flood more severe than the design criteria.

Comment
Number

3.e.4

p. 3.1.116

Table 3.1.11 purports to give estimates of resources needed for construction and operation of waste repositories in various geologic formation for different fuel cycle options. It also compares effluents for the various options. However, no basis for any of the numbers listed is given. The basis for such estimates should be included.

3.e.5

p. 3.1.118

Table 3.1.12 presents total quantities of effluents released to the atmosphere during construction and operation of a geologic repository. The potential effects of these effluents on ecosystems should be evaluated.

3.e.6

p. 3.1.120

There is little or no discussion of the potential hydrologic implications of repository construction and operation. For example, what would be the effects on surface drainage and downstream water quality of excavated material stored on the surface? Would the material be laid out on level surfaces, would low spots be filled in, would streams be diverted or dammed? What would happen during heavy rain and/or floods? Where would water needed for construction/operation be obtained? A description of a typical site, its construction and the hydrologic and water use impacts is needed.

3.e.7

p. 3.1.120

A more detailed discussion of the ultimate disposal of excavated material is needed. In some ways this problem is analogous to the disposal of dredged material. The volumes (tens of millions of cubic yards) are similar to those involved in large dredging operations. It cannot be dismissed out of hand without more detailed discussion.

3.e.8

p. 3.1.120

It is stated that the regional population dose for a geological repository during construction and operation is 100 man-rem. However, no reference is given to the basis for this estimate. For example, how much radon is estimated to be released during construction and operation at the repository.

Comment
Number

e.9 p. 3.1.123
Provide justification for all the assertions in the discussion of a tornado strike. Specifically: the dimensions of the salt pile, the size of the pieces, the probability of the tornado, its maximum wind speed, the amount of material removed and the resultant concentration in air.

In addition, no reason is given for discussing this accident. Is it the worst nonradiological accident possible, is it the only one considered, or is there another reason for its choice? What about other accidents? No conclusions are presented. Should measures be taken to protect salt piles from tornados? Has a cost-benefit analysis been made?

3.e.10 p. 3.1.126-132, 3.1.179-181, 3.1.184-186, 3.1.193-195, 3.1.200
GEIS is characterized as generic and not site specific (page 3.1.98). The document further states that the ability to identify socioeconomic impacts increases as one proceeds from a generic to a site-specific situation. However, a model was employed which provided and compared very specific social service demands anticipated for each of the reference sites. It is unclear why the analysis, which used actual site specific population, employment, education and housing information to estimate social service demands, did not relate the demands to existing capacities to indicate net impacts.

The reference sites are compared and the comparison reveals a range of different conditions and anticipated social service demands. Are these reference sites being presented as being representative of sites to be found in the Southeast, Southwest and Midwest areas of the country? How much variability can one expect to find among sites within the geographical boundaries of each of the above areas (Southeast, Southwest and Midwest)? If large differences are expected within each of the geographical areas, to what use is the reviewer to put comparative information presented in GEIS?

Comment
Number

While a considerable amount of useful information is presented in terms of manpower needs and expected social service demands for the three reference sites, the demands are not related to the infrastructure capacities of the expected impacted communities to ascertain net impacts. The subjects of compensation, payments in lieu of taxes, and mitigation in general, need considerably more development.

3.e.11 pp. 3.1.179, 183, 188, 203
No basis for any of the values of resources committed shown in Tables 3.1.55, 62, 65, 78 are given. In addition, no units are given for operational water use, concrete, propane and electricity in Table 3.1.55.

3.e.12 p. 3.1.223
The third paragraph suggests that water use will not be a problem. The basis for the statement was the assumption that the facilities could all be located near the "R" river, which had adequate flow. However, the statement should recognize that water use could be a significant environmental impact for a repository which cannot be located near a convenient water source.

The resource commitments listed include annual water use for the once-through fuel cycle option. The total annual use is about 1% of the annual mean flow of the "R" River, a small amount when water is plentiful. However, in the semi-arid west where river flows can be less than 100 cfs (one-fiftieth that of the R River) and where water is fully allocated, this is a significant amount of surface water use.

3.e.13 p. 3.1.226
The 2nd paragraph states that there will be 10's of millions of tons of salt "whose final disposition is yet undecided." It is not clear where this fits into the analysis, or if it is taken into account, where in comparing environmental impacts in the various geological media does this occur.

Comment
Number

. f Long-Term Effects of Repository Construction and Operation

3. f. 1 pp. 3.1.5, 3.1.6, 3.1.19

One area of serious concern which appears to be neglected is the effect of repository construction and operation and the thermal effects of the emplaced waste on the effectiveness of the geohydrologic barriers to long-term transport of radioactive materials from the repository. The following examples are cited:

Construction is likely to increase hydraulic conductivity of the rock mass. There is no evidence presented in GEIS to show that such factors have been considered. This is a serious deficiency.

Rock fractures, joints and fissures are potential paths for increased groundwater flow. Mine construction and testing may induce local fracture conditions that may or may not be identified in sample permeability testing. However, the in situ extent of fractures, joints, and fissures could produce increased groundwater flow in other than direct downgradient directions. Have such factors been considered and what conclusions have been drawn?

3. f. 2 pp. 3.1.9, 3.1.13, 3.1.14

pp. 7.2.15, 7.2.18, 7.2.22, 7.2.26; DOE/ET-0028

Permeabilities of granite and basalt, while low, are not nil. If they were, the repositories in granite and basalt could be located a few meters beneath the weathered layer. There seems to be no appreciation that values of permeability determined in the laboratory differ quite frequently from effective (rock mass) permeability by several orders of magnitude.

3. f. 3 p. 3.1.24

Thermal uplift around the repository is expected as a result of the thermal loading. This may increase effective hydraulic conductivities of the host rock and may even result in the creation of a flow path between overlying aquifers and the repository.

Comment
Number

. f. 4 p. 3.1.32

The first paragraph does not reflect fissure and joint permeability differences, and induced characteristics due to construction.

3. f. 5 p. 3.1.124

What the GEIS is really discussing is the creation of flowpaths by creating fractures or opening fractures that already exist. The question then is how does the predictive model treat the flow of liquids and transport of dissolved radionuclides through fractures? Both flow and transport could be significantly different in fractures than in porous media. We know that retardation is less and, also, that retardation is the most important attenuation mechanism that has been modeled. The term "thermally induced permeability" does not convey the difference between porous flow and fracture flow.

3. f. 6 pp. 3.1.228, 3.3.22, 3.3.27, 3.3.30
pp. 7.4.6, DOE/ET-0028

The general impression conveyed regarding the sealing of shafts, bore holes and canister holes into an underground repository is that no significant problems of leakage are expected. This contrasts with the discussion of the same activities associated with disposal of wastes by the very deep hole concept. If anything, the maintenance of the integrity of shaft seals, room seals and canister seals in an underground repository (particularly in salt) would be expected to pose significantly greater problems than in deep hole disposal.

3. f. 7 p. 7.3.5; DOE/ET-0028

The unit cell used to analyze the maximum temperature of the waste canister is described on page 7.3.5. It has a sleeve and an air gap. At the bottom on page 7.4.6 it is stated: "After the readily retrievable period, use of sleeves in emplacement holes is discontinued and holes are backfilled with crushed rock after canister placement." Emplacement of waste canisters without the sleeves does not appear to be considered in the thermal analysis.

Comment Number

- 3.f.8 p. 7.3.12; DOE/ET-0028
Linear thermomechanical analysis is applied to a repository in salt. Such an analysis can result in significant error in predicting thermomechanical effects (see discussion under comments on retrievability). Even with this analysis a surface uplift up to 1.5m is predicted. The important question not addressed in the GEIS is what effect will this have on shaft and borehole seals, thermally driven convection and breccia pipe formation?
- 3.f.9 General Comment
The GEIS does not address the important and complex problem of groundwater mass transport and how it is affected by joints and fractures. Rock will fracture and a series of joints will be created or opened in the surrounding rock as a result of mining of the repository. Effective permeabilities (hydraulic conductivities) will be increased. The effect of these processes on long-term repository performance need to be addressed.
- 3.g Long-term Radiological Effects - Environmental Transport
- 3.g.1 p. 1.20
Some of the numerical values on page 1.20 (e.g., maximum individual dose) cannot be traced to Section 3.1.5.
- 3.g.2 p. 3.1.66
Describe how the estimate of 1×10^{-6} for the "annual fatalities estimated due to isolated waste" is determined. Specifically, explain and justify the use of the "annual transfer probability for an atom of radium to enter the body from the geosphere."
- 3.g.3 General Comment
Define "health effects" and assumptions for "translating" 1.8×10^8 man-rem into 1.8×10^4 to 1.4×10^5 health effects.

Comment Number

- 3.g.4 General Comment
Although the section on radiological models (Appendices D & F) indicates that all pathways were considered, the contribution of various pathways to the total dose is not given in the document. Additional information on the radiological analysis for scenarios (e.g., source terms, concentrations of nuclides for different locations, solubility classifications of particulates, etc.) would help document the major conclusions concerning radiological impacts.
- 3.g.5 General Comment
Appendix I to the EIS and Appendix G to DOE/ET-0029 present impacts at time periods of 10^2 yr., 10^5 yr. and 10^6 yr. Sometimes 10^4 years is discussed. Since preliminary versions of the EPA standard for high-level waste specifically reference the 10^4 year period, it would be prudent to present cumulative dose calculations for this time period for all cases studied.
- 3.g.6 General Comment
The several appendices which support the long-term impact assessment need to be coordinated so that their results are directly comparable. Some cumulative doses are for 50 yr., some for 70 yr. Different times are referenced. The total picture is confusing and leaves many questions about the internal consistency of the supporting calculations.
- 3.h Long-Term Radiological Effects - Geology/Hydrology
- 3.h.1 p. 3.1.2
The four climatic factors listed to be considered in assessing the long-term isolation of waste are not sufficient. Precipitation patterns (temporal and spatial) and man induced changes must also be considered.
- 3.h.2 p. 3.1.5
It is stated that "The major mechanisms related to nuclide transport through the disposal media are thermal convection, diffusion and dispersion, sorption, and radioactive decay." Also important can be the advection of nuclides with the local groundwater flow.

Comment
Number3.h.3 p. 3.1.5

The definition of convection is incorrect. Convection signifies the transport of a contaminant by a moving fluid. Thermal differences may produce fluid motions, and thermally driven convection must be considered in the analysis of a radioactive waste repository. The usual driving force for groundwater flow is the head gradient (where the head is due to elevation and pressure). The velocity and direction of flow are governed by a combination of fluid properties, rock properties and head gradients.

3.h.4 p. 3.1.6

Line 2: To group geologic materials into two categories, either aquifers or aquitards is misleading. A whole continuum of both permeability and porosity exists which can describe an aquifer (high permeability and high porosity), aquitard (low permeability), and aquiclude (very low permeability but may contain appreciable porosity) also known as an impervious horizon or an aquifuge (very low permeability and very low porosity). The local site conditions generally determine how you would classify the hydrostratigraphic unit since these terms are often relative.

3.h.5 p. 3.1.6

Line 4: A discussion is needed of piézometric levels, leakage between confined, unconfined, and various combination hydrostratigraphic units, and how a unit may change from a phreatic, to a confined, leaky, or artesian aquifer.

3.h.6 p. 3.1.6

A discussion of steady state versus transient flow conditions and their implications on hydrostratigraphic unit storage is needed. The variability of parameters governed by the matrix plus secondary features such as faults, joints, structure, and alterations also need discussion.

Comment
Number3.h.7 p. 3.1.23

It is stated that interior drainage is a favorable hydrologic characteristic in selecting a burial site. An example given is the Great Basin of Nevada and Utah. However, one characteristic of interior drainage is that during wet climatic periods they can become almost completely water-covered. This has happened in the recent geologic past in the Great Basin. Consequences of the potential for such drastic changes in the surface and subsurface water regime should be more carefully investigated before asserting that interior drainage is favorable. Interior drainage is again favorably mentioned on page 3.1.27.

3.h.8 p. 3.1.8

Because of the complexity and nature of deep geologic and hydrologic investigations, simple analysis using permeability, porosity, and hydraulic gradients are not sufficient. Appropriate parameters for evaluation of hydrologic regimes are:

fluid properties

density

compressibility

thermal expansion/heat capacity

viscosity

matrix properties

longitudinal and transverse dispersivities

vertical permeability

density

compressibility

storage and leakage factors along with permeability and porosity.

In addition one needs to assess the difficulty in determining these parameters, their uncertainties and extent to which and time required for a hydrologic model to be validated and calibrated. The above items may contribute significantly to uncertainty in predicting future safe performance. They could also impact the date for initial emplacement.

Comment
Number

3.h.9

pp. 3.1.24, 3.1.33, 3.1.235

Several comments have been made about the "self-healing" properties of salt:

p. 3.1.33 - "Fractures tend to self-heal, thus reducing...water ingress..."

p. 3.1.24 - "A key problem will be preservation of low permeability. Preliminary thermal loading analyses indicated that tensile forces will be induced near the outer margins of the repository. Thus, thermal expansion could create potential pathways for work migration by fracturing or by opening pre-existing fractures. For salt strata this is not a problem; salt is expected to deform plastically and heal internal fractures. However, the problem is that if the surrounding strata were breached by fracturing, salt could be vulnerable to rapid solution by groundwater. Therefore, it appears that thermally induced permeability will be an important consideration for all host rock media."

p. 3.1.235 - "...generally accepted...salt tends to heal any opening"

It may not be realistic to depend on this "self-healing behavior" to produce an impermeable seal around the repository. The repository design should consider worst case behavior. Worst case behavior would be the opening of thermally or mechanically induced fractures around the repository to water flow from an overlying aquifer. The water under greater pressure due to depth could keep the fractures open and increase the dimensions of the fractures as a result of the flow.

3.h.10

p. 3.1.26

The great deficiency in the hydrogeologic data base is actual field studies and methods for obtaining rock dispersivities. Also lacking are in-situ sorption studies for a variety of geologic, hydrologic and geochemical environments. (Note: The Canadians are doing work in this area at the Chalk River Nuclear Laboratories (CRNL)).

Comment
Number

.h.11

p. 3.1.33

The GEIS states that "Mines in Canadian Shield Granite appear to be tight and free from circulating groundwater below depths of about 3000 ft." We recognize that granite has a low hydraulic conductivity and that seepage rates are low enough to appear negligible by visual inspection. However, it is likely that groundwater inflow into operating mines is evaporated by ventilation airflow. In the long time frame of a repository, this inflow is expected to be significant.

3.h.12

p. 3.1.41

Groundwater nuclide transport is not included among the issues needed to be resolved to determine post-operational impact of the repository (p. 3.1.41). On page 3.1.48 and 49 it states that groundwater movements that are insignificant over the short term could be a problem where considered over the long term. Groundwater movement in a salt repository is considered to be negligible.

Table 3.1.49 on page 3.1.164 implies that there could be an unacceptable 50 year body dose as a result of the groundwater transport of radionuclides by the year 2050. Is this in contradiction to other passages discounting the effects of mass transport?

3.h.13

p. 3.1.41

Operational difficulties which may prevent sealing the repository have not been discussed. It is difficult to see how one could do an adequate job of either backfilling or retrieving if a repository becomes flooded. The point to emphasize is that operational problems may impact long-term performance. The effects of contaminating the repository in an accident, which may affect both occupational safety and long-term performance, are not addressed.

3.h.14

p. 3.1.47 to 3.1.76

For a repository in salt, a discussion of brine migration is missing. There was no mention of the possibility that sorption of the effluent of a

Comment
Number

salt repository may not be the same as for other media, due, for example, to competition for sorption sites by Na^{++} , Mg^{++} , and Ca^{++} .

3.h.15 p. 3.1.67

Uncertainties and the method for determining them should be consistently included with probability and consequence estimates. Although there is some discussion of uncertainties in isolated cases, they are usually not included with point values, e.g., the probability of faulting through the repository is estimated at 4×10^{-11} year (pg. 3.1.67) with no indication of associated uncertainties.

3.h.16 p. 3.1.98 and Appendix I

It is stated that "...methods and detailed results for groundwater transport of radionuclides are presented in Appendix I." However, Appendix I contains no detailed discussion of groundwater transport models. That appendix is primarily a discussion of radiological consequences of leaching of waste in a repository. The hydrologic assumptions stated and presumably used in the modeling (which is not discussed) are simple (e.g., constant velocity). There is no discussion of the effects of different hydrologic characteristics, i.e., no sensitivity analysis.

3.h.17 p. 3.1.120 to 3.1.123

The discussion in GEIS under "routine releases of radioactive materials" does not address the problem of radionuclide contamination of groundwater and run-off water. This could happen as a result of accidents, clean-up operations in storage rooms, decontamination operations during the retrieval cycle, etc.

In the section titled "Ecological Effects" seepage and water inflow from overlying strata for repositories in granite and in shale are discussed. The estimated inflow of water in a granite repository ranges from 550 to 1550 m^3/day . The estimated maximum inflow during the last stages of operation will range from about 3,800 to 19,000 m^3/day (50000 gpd). There appear to be two implications by omission from the discussion:

Comment
Number

- o No continued water inflow is expected in the repositories in granite and in shale after the last stage of operation.
- o No water inflow is expected in the repositories in salt and in basalt.

The generic stratigraphy for salt includes possible aquifers overlying the salt bed. An area of uncertainty in state-of-the-art technology is whether the effects of mining a repository in salt and of the thermal loading are such as to create fractures that would connect the aquifer bed to the repository. TM-36/21 (p. c-1) discounts this in assuming that the permeability for salt remains at zero. No justification is provided.

3.h.18 p. 3.1.136

Justification is needed for the stated maximum surface temperature rise and uplifts.

3.h.19 p. 3.1.148-3.1.155

Discuss the reasons for the choice of $2.8\text{m}^3/\text{sec}$ (100 cfs) for water flow through the breached repository. Identify the flow rate of hypothetical river "R" used in transport and dilution calculations.

3.h.20 p. 3.1.158

Provide a reference for ten dilution factors given and discuss the cause of the 50 fold differences shown.

3.h.21 p. F.3, Appendix F

The hydrology of the hypothetical site is presented with no explanation or discussion of its appropriateness for general sites. No discussion of other hydrologies is given. Considering the great length of discussion that is given throughout the document to effects of comparatively small changes in the characteristics of the waste, an apparent lack of appreciation of the effects of the sites hydrologic characteristics is manifested by this treatment.

Comment
Number3.h.22 Appendix I

Appendix I discusses the possibility of release of radionuclides to the biosphere through groundwater mass transport. The impression given is that container life will be about 1000 years and that no significant release is expected for one million years. This is in apparent contradiction to results given in TM-36/21 (p. xiv, 8-5 and 8-6). What is the expected rate of corrosion of the canister and the sleeve in salt brine or in fresh water? What are the values (or ranges) of effective hydraulic conductivity, porosity, retardation factors and hydraulic gradients of the rock mass surround the repository that were used to obtain Tables I.1 to I.12?

3.h.23 Y/OWI/TM-36/21

Pages 4-7 assume that the effective vertical permeability of basalt between the repository level and the alluvium near the surface (a thickness of 600 feet) is 5×10^{-8} cm/s resulting in a downward flow through this layer into the repository of approximately 150 gpm (216000 gpd). In addition a maximum upward flow of 230 gpm into the repository is calculated.

The GEIS should address and discuss the following with regard to radionuclide transport: Are repositories in granite, basalt, salt and shale expected to have any water inflow after the last stages of operation?

3.h.24 Y/OWI/TM-36/21

The results of simplified calculations given in Y/OWI/TM-36/21 show ^{99}Tc exceeding acceptable concentrations 3 miles from the center of the repository 400-600 years after recharge. To quote from page 8-5: " ^{99}Tc , due to its long half life and unity retardation coefficient exists in all layers of the generic stratigraphic columns studies (shale, granite and basalt) at concentrations near or equal to the source activities. The maximum source activity for ^{99}Tc used in this study is approximately 0.2-0.3uCi/ml (section 7.0) which is at least 10^3 times greater than an acceptable level. The first arrival of ^{99}Tc occurs in the near surface layers between 400-600 years after repository decommissioning and resaturation and at

Comment
Number

concentrations near or equal to that of the repository source activity." This would appear to indicate unacceptable repository performance. An explanation should be given of how this will be remedied or why this analysis is not believed to indicate a problem.

3.h.25 Y/OWI/TM-36/21

Y/OWI/TM-36/21 addresses only three host rock media - granite, basalt and shale. No basis for the apparent conclusion that groundwater movement in salt is negligible has been presented in either GEIS or in TM-36. Note also that the permeabilities of granite and basalt presented in the GEIS (Table 3.1.1, p. 3.1.9) are nil and therefore the repositories in granite and basalt could presumably be located at depths significantly less than salt and shale.

3.h.26 General Comment

Measures of performance used in the GEIS and its supporting documents make it difficult to judge statements that claim "no deleterious effects." For example:

1. Dose received by maximum individual. This seems to be someone using a water supply separated by 10 miles of porous flow from the repository. Note that fracture flow with its lower retardation factor is not considered.
2. Concentration at 3 miles from boundary. This was used in TM-36 volume 21. In this case, Tc-99 occurs near the surface at 400-600 years and exceeds maximum permissible concentrations by one thousand (Tm-36/21 pgs. xiv, 8.5-8.6).

3.h.27 General Comment

One of the assumptions that makes mined geologic disposal feasible is that radioactive sources placed in a hydrologic environment with slow-moving groundwater will take long periods of time to be transported to the biosphere. Furthermore, retardation effects will slow down (relative to groundwater velocity) the movement of certain species. This basic characteristic is common to all forms of geologic disposal.

Comment
Number

The GEIS and its supporting documents fail to analyze flowpaths other than porous flow through intact media. The possible creation of high-velocity flow paths by mining operations or fractures created by the thermomechanical response of the rock mass are not considered. Fracture flow driven by thermal convection deserves more attention than meteorite impact or nuclear war as mechanisms for establishing communication between the repository and the biosphere.

3.i Long Term Radiological Effects - Accident Analysis3.i.1 p. 1.16

In the definition of risk, "magnitude of the loss" is better expressed as "consequences of the event." This will also make the definition of risk consistent with that used in footnote e to Table 1.4 and the footnote on page 1.21.

3.i.2 pp. 1.16 and 1.20

We note that a risk assessment requires the identification of a broad spectrum of event probabilities and consequences. It is not limited to worst case consequence assessments as is indicated in Tables 1.3 and 1.4.

3.i.3 p. 1.19

A credible event missing from the discussion is the possibility of a water well drilled into adjoining hydrostratigraphic units that could disrupt regional flowlines and equipotentials such that radionuclide migration may be enhanced. Leakage through overlying aquitards into more permeable units could significantly speed the movement of radionuclides to the biosphere. The pumping well in this scenario would not be pulling radionuclides directly into its cone of depression since most water wells are not at that depth nor would the repository be located in a productive aquifer of potable grade water. Further, the discussion on solution mining and the missing scenario on deep drilling activities such as natural gas and oil exploration ignore the potential for groundwater hydraulic and pollution effects.

Comment
Number3.i.4 p. 1.20

Table 1.4, Item 1: Although the person closest to the repository will be killed, there still exists a maximum individual who receives the largest dose as a result of the release.

3.i.5 p. 1.20

In Item 3 of Table 1.4, the regional natural radiation dose is calculated for 3 generations. In Item 2, doses are calculated for only 1 generation (70 yr. total body) resulting in an inequitable basis for comparison.

3.i.6 p. 1.20

Table 1.4 - (a) The potential for a dose due to airborne dispersion caused by a meteorite impact does not appear to have been considered, (b) the units of "Health Effects," e.g., acute fatalities, morbidities should be defined, (c) the units of "Risk," e.g., total health effects, health effects per year should be defined, and (d) a description of how "accident probabilities" were arrived at and an associated uncertainty should be presented, e.g., both the probability for meteorite impact and the probability for fault fracture and flooding were given as 3×10^{-13} . Including uncertainty in the estimates of probability is also important since point estimates of probabilities as low as 10^{-13} , are difficult to justify when little data is available.

3.i.7 p. 1.21

Artifacts survive but if they have value as collector's items or useable resources (e.g., high grade steel) there may be considerable motivation to move or destroy them. The problem is not only one of designing a marker that will last and be understandable but also one that will stay put without being defaced.

3.i.8 p. 3.1.2

Only erosion is mentioned as a hazard associated with glaciation. Omitted are faulting and deformation well below the eroded rock/soil surface. These potential hazards should also be considered when evaluating the effects of glaciation.

Comment
Number

8. i. 9 p. 3.1.2
Considering the multitude of variables and unknowns, it would seem extremely difficult to predict the lower depth of glacial erosion at any particular site with any degree of certainty. A more acceptable approach would seem to be that, if the decision has been made to seriously consider a repository within a previously glaciated area that the repository designer would simply assume surficial erosion (deposition and various deformation/faulting features) to occur within, say the upper 65 to 100 m of the surface. Other than probably uniform crustal depression, a repository located at the 500-600 m depth should be relatively unaffected by direct glacial processes. It would seem to be overly-conservative to assume that a postulated future glacial front would advance beyond the areas formerly occupied by continental glaciers.
3. i. 10 p. 3.1.65
It is stated that "containment times of 500 years are the most important." However, on page 3.1.59 it was stated that a significant release" could occur at 1000 years and on page 3.1.64 it stated that after "700 years, the radioactivity in the repository poses a greatly reduced threat." Some consistency should exist in the document for the period of concern and basis for arriving at this time should be clearly delineated.
3. i. 11 p. 3.1.67
Table 3.1.3 - It is stated that the Poisson process is used to model the occurrence of geologic events, based on past observation. It is not clear, however, whether this table presents the probability that one event occurs for the "interval" of concern or, more properly, that one or more event occurs during this period. From $P(x) = \frac{e^{-gQ} (gQ)^x}{x!}$, the probability of one or more events occurring is $(1 - \text{the probability of zero occurrences}) = (1 - P(0)) = 1 - e^{-gQ}$. This formulation, however, produces somewhat higher probabilities than those listed in Table 3.1.3, e.g., for the "number of occurrence years" equal to 10^6 years, and an "interval" equal to 10^4 year, the probability that one or more geologic event occurs is 9.95×10^{-3} as compared to 6.9×10^{-3} . Thus, more explanation of the probabilities in Table 3.1.3 is needed.

Comment
Number

- . i. 12 p. 3.1.125
Individual and population doses, as well as health effects, are calculated and presented in the GEIS for certain postulated accidents. The potential decontamination cost and property damage associated with the same postulated accidents should also be evaluated.
3. i. 13 p. 3.1.125
Uncertainties in doses predicted by models is misleading. The discussion should indicate the uncertainty to be expected when a critical parameter has a range of values such as the magnitude of earthquakes, floods, etc.
3. i. 14 p. 3.1.125
The discussion in the section entitled "Land Use and Transportation Considerations" focused on some possible land use conflicts and refers the reader to a body of literature, some of which is described as speculative. It would be useful for the GEIS to summarize this information and to present it for review.
3. i. 15 pp. 3.1.136-172
In this section several scenarios resulting in the release to the biosphere of large amounts of radioactivity are postulated. Because of the generic nature of the repositories and the lack of specific data needed in the calculation, many of the parameters controlling the physical transport of the radionuclides are not even known to order of magnitude certainty. The resulting dilutions that are used in the dose models have even larger error bands. Therefore, breaking down the resulting doses by reprocessing procedure and rock type makes little sense, when the differences between them are much less than the error band due to transport-dose modeling.
3. i. 16 p. 3.1.136
Section 3.1.5.2 is entitled, "Potential Impacts Associated with Repository Wastes in the Long-Term." Although this section gives population doses due to different accident scenerios, it does not discuss the problem of land contamination due to these accidents.

Comment
Number3.i.17 p. 3.1.137

The section on long-term impacts is devoted entirely to accidents that may breach the repository, most of which are presented as being so improbable that they are unlikely to ever occur. There is no discussion presented of expected long-term impact. If the facility is sited, filled and sealed according to plan, what will the long-term consequences of this action be in the absence of unlikely accidents? This question is discussed partially in Appendix I but the discussions are not presented in the text of the GEIS as projected impacts of the action.

3.i.18 p. 3.1.137

Releases are estimated for four hypothetical accident sequences. The numbers associated with the releases are presented by the GEIS as "what if" calculations, without discussion of why these sequences are important except to say that they are "believed most representative" of release events. How these events were chosen and why they are believed to be representative and to bound the impact of long-term consequences should be discussed.

3.i.19 p. 3.1.138

References relevant to this discussion and not cited include:

1. K.A. Solomon, R.C. Erdmann and D. Okrent, "Estimate of Hazards of a Nuclear Reactor from the Random Impact of Meteorities," Nucl. Technical, 25. 68 (1975).
2. K.A. Solomon, R.C. Erdmann, T.E. Hicks and D. Okrent, "Estimates of the Hazards to a Nuclear Reactor from Random Impact of Meteorities," USCL-NEG-7426, University of California at Los Angeles (March 1979).

3.i.20 p. 3.1.150 to 3.1.155

"The annual doses to a maximum individual associated with the breach of a salt repository are three to ten times the permissible annual dose for occupational exposures ... Thus the calculated doses and consequences

Comment
Number

seem most unlikely to occur in practice the calculated number of health effects attributable to this accident would range from 1×10^4 to 3×10^5 ."

GEIS goes on to multiply these figures by 1/100 as the probability of failure of waste containment and by 4×10^{-11} /yr as the probability of a new fault intersecting the repository to arrive at insignificant risk levels. The probability of an existing fault becoming permeable should also be considered.

3.j Research and Development3.j.1 General Comment

It would seem advisable, if not already considered, to gather information regarding the long-term stability of boreholes, wells, and other deep rock penetrations in regions considered favorable for repository location. These observations can provide additional clues on assessing the stability of the repository location. This would be useful in assessing the host media as well as that of the overlying and underlying formations especially when considering the Very Deep Hole concept of waste isolation. Perturbations of the earth's near-surface are readily detectable in both cased and uncased holes through sheared, ruptured, and squeezed boreholes and casings.

3.j.2 p. 3.1.237

There is no discussion of research needs in the hydrologic transport aspects of geologic disposal. Of prime importance are the chemical and thermal interactions involving dissolved wastes and the natural rock.

3.k General3.k.1 p. 1.3

The underground firing of nuclear explosives results in the formation of vitrified debris, due to the solidification of molten and vaporized rock. Thousands of tons of such vitrified debris have been in place for periods of up to 25 years, mostly in tuff at the Nevada Test Site, but also in

Comment
Number

granite, shale (Gas Buggy), and salt. This experience bears directly upon the proposed long-term storage of vitrified high level waste, and should be discussed.

3.k.2 p. 1.3

The need for additional in situ testing to obtain site specific information should be stressed in the GEIS. For example, acceptability of a shale as the host media at one location does not imply that a shale at another location is necessarily acceptable since nonlithologic parameters such as tectonic setting, in situ stresses, hydrology, and other variables are undoubtedly different.

3.k.3 p. 1.12

Why are salt, basalt, granite and shale considered to be representative of all geologic media? Some explanation should be given.

3.k.4 p. 3.1.8

Rock structure and texture are not interchangeable terms. A glossary of geologic terms used in the GEIS, such as structure, texture, lithology, bedding, and joint may eliminate confusion concerning the usage of standard terms and should be provided.

3.k.5 p. 3.1.8

The confining earth pressures whose release cause joints should be characterized. For example, glacial retreat and thermal contraction should be named as causes of jointing in rock.

3.k.6 p. 3.1.8

Salt domes may deform overlying strata without penetrating them. Therefore, "deform" should be substituted for "penetrate" in the 6th sentence.

3.k.7 p. 3.1.9

The statement, "...the water incorporated in them (salt beds) was trapped when the beds were formed and does not migrate," is erroneous. Fluid inclusions in salt migrate along thermal gradients.

Comment
NumberB.k.8 p. 3.1.9

The statement, "Joints can be ..." is too vague. Are joints usually, often, or seldom anhydrite-filled, near vertical, unopen, moderately spaced, and generally extensive?

3.k.9 p. 3.1.11

Does "hard" refer to hardness (as in scratch test) or strength? Geologic terms such as strength and hardness should be used in accordance with their standard definitions.

3.k.10 p. 3.1.21

References for the statement "...shaking ...due ...to earthquakes is not expected to have serious effects on the repository at depth..." should be provided.

3.k.11 p. 3.1.23

Groundwater may constitute the major potable water supply of many western states. This aspect of groundwater importance should be addressed.

Comment
Number

4. ALTERNATIVE DISPOSAL CONCEPTS

4. a Geologic Emplacement Following Chemical Resynthesis

4.a.1 Chemical resynthesis is not an alternative waste disposal concept but rather an alternative waste form which would be a candidate for a number of disposal alternatives presented in this document. The designs of deep geologic repositories place major (if not total) reliance for containment of radionuclides on the surrounding geology (See Section 3.1.1). Reliance on the waste form itself and its packaging to prevent radionuclide release over the long term has not received intense emphasis. For example, Section 3.1.4.2 points out that the reference solidification process for conventional geologic disposal is conversion to glass, as the alternative waste forms are less well developed.

4. b Very Deep Hole Concept

4.b.1 On page 3.3.1

It is stated:

"In summary, the deep hole concept cannot be evaluated as a nuclear waste alternative without more information on the deep groundwater system, rock strength under increased temperatures and pressures due to decay of wastes, and the sealing of the holes over long periods of time."

These are three areas that have also been identified under the research and development needs section (Section 3.1.6) for Conventional Geologic Disposal.

- a. Why does the evaluation of deep hole disposal as an alternative depend on obtaining this information, while it is taken for granted that conventional Geologic Disposal is a viable alternative?
- b. If this information is obtained for conventional geologic disposal, would it apply to deep hole disposal?

Comment
Number

4.b.2 Some discussion of retrievability from deep holes should be provided.

4.b.3 p. 3.3.33

It is stated that, "It will be necessary to locate sites in strong, unfractured rock of low water content." This will exclude such media as shale and salt because of strength, and most other media because of fracturing. Why hasn't this same site selection criterion been applied to conventional geologic disposal?

4.b.4 p. 3.3.33

The section on the thermomechanical behavior of rocks does not acknowledge that a significant body of information has been published on studies of hydrothermal alteration of natural rock bodies. The time, temperature, and the nature of ion migration in hydrothermally altered rocks has been studied for years by igneous/metamorphic petrographers, geochemists and mining companies.

4.b.5 p. 3.3.37

The citation for Reference 27 is inadequate. Provide information whereby Mr./Ms. Stevens can be contacted.

4. c The Rock Melting Concept

4.c.1 General

The Rock Melt Concept discussed in Section 3.4 assumes that the cavity is loaded over a period of years. This prolonged loading time has at least two disadvantages. First, the physical integrity of access and venting shafts must be maintained for the duration of the loading. Second, the cooling water itself will be contaminated and must be carefully contained and eventually the contamination must be disposed of as yet another waste.

Another loading scheme should be considered. The waste could be stored at the surface until the full load for the cavity has been accumulated. The waste could then be rapidly loaded into the cavity and the cavity quickly sealed.

Comment
Number

It appears that the quick loading of the cavity is a practical alternative to the prolonged loading suggested in the GEIS. Further variations should also be considered, such as the use of an array of cavities (a few to maybe 10's of cavities). This would reduce the loading rate (in the case of the quick load) and distribute the heat load over a large volume.

4.c.2 General

The treatment of "Rock Melt" in the GEIS misleads the reader as to the depth of investigation which has been completed. For example in the first paragraph on page 3.4.4 of the GEIS, it is stated: "The concept has been assessed and reviewed (4,5) and preliminary laboratory scale investigations have been performed (6,7)." The workshop referred to as Reference 5, as productive as it may have been, fell far short of assessing "Rock Melt." The laboratory scale investigations were designed to study the descent of solid containers by rock melting, not the molten cavity concept.

4.c.3 p. 1.25

The introductory writeup on the rock melting concept does not present the disadvantages for this alternative, which were presented for the very deep hole concept, sub-seabed geologic disposal, etc. Equal treatment of all alternatives should be demonstrated in the final EIS.

4.c.4 p. 3.4.5

It is stated that retrieval of waste following emplacement would be difficult. This is understated, and not adequately addressed.

4.c.5 p. 3.4.6

It is stated that the consequences of seismic activity appear minimal with proper facility design. Discuss the effects of seismic activity on surface facilities supplying cooling water and cleaning up the steam, and on the reliable supply of cooling water to the waste.

Comment
Number4.c.6 p. 3.4.10

Figure 3.4.4 does not present the temperature profiles that are necessary to completely characterize the extent and duration of the thermal load on the host media. The maximum increase in temperature at the earth's surface can occur hundreds of thousands of years later than shown. (Numerical models can be very costly to run for long times and distances required, however an analytic model is available. See Reference 3 of Appendix C of TID-28818 (Draft), "Subgroup Report on Alternative Technology Strategies for the Isolation of Nuclear Waste.")

4.c.7 p. 3.4.17

The post sealing period environmental effects are assumed to be "the same for (nonsalt) conventional and Rock Melt repositories." The basis for this assumption should be given. If the thermal barrier effect protects the HLW from groundwater leaching for possibly a few thousand years, might not the post sealing performance be superior to that for conventional geologic disposal?

4.c.8

A shortcoming of the description of the rock melt alternative is that no mention is made of the need for or availability of the water that's necessary for this alternative. Provide an estimate and discussion of the water requirements.

4.c.9

In the event that the cooling system for the waste fails while still needed, it will be very difficult to repair because of its proximity to the waste.

Information on the possible failure of the cooling system, mitigative actions, and environmental impacts should be provided.

4.d Island Disposal4.d.1 Section 3.5

The discussion in Section 3.5 indicates that two options for island disposal are being seriously considered. One option is disposal in

Comment
Number

oceanic islands for which relatively long sea voyages for transporting the radioactive wastes will be necessary. The other option is disposal in continental islands. For this option, the transport time at sea is small with the possibility of using a ferry-type transport system, facilities at the embarkation and receiving port could be simplified. Table 4.2.1 indicates that an offshore continental island has been chosen as the reference system. The two options should continue to be treated separately and additional information concerning environmental impacts and accident risks be developed for both options. Note that although the offshore continental island option appears to be the option with the least transportation environmental impact, it also has associated with it the least benefits. Section 3.5.1 states that the concept of the island disposal is being considered because of the benefits derived from this disposal option. Benefits such as location in a separate hydrogeological zone, seawater dilution of radioactive leaks, enhanced security of a remote location, and a site with international jurisdictional status would all be minimized if the offshore continental island option is chosen. It is important to continue to explore both options with the ultimate choice being left to a risk-benefit analysis after more complete information is developed.

4.d.2 Section 3.5

The ability to dewater a site is an extremely important site characteristic. Dewatering with the attendant equipment may impose such an economic burden that an otherwise suitable site may be ultimately rejected. The dewatering problem may, in the end, result in the rejection of the island arc and oceanic island locations. In addition, the retrievability of waste placed in any island watery environment, particularly salt water, is questionable considering the effects of corrosion on dewatering equipment.

4.d.3 p. 1.25

Section 1.3.5 states that "Salt deposits are unlikely to be available at island sites; the most probable disposal formation (sic) is crystalline

Comment
Number

rock." From this statement one would conclude that crystalline rock was the most common rock type exposed on islands. This is not the case, e.g., the Antilles, the Japanese and Philippine archipelagos, New Guinea, Bikini, Bermuda, etc.

4.d.4 p. 1.26

The statement on line 9, that island arcs are highly active seismically and volcanically is not necessarily correct as there are tectonically inactive island arcs.

4.d.5 p. 3.5.1 (also on page 3.5.5)

The assumption of a "practically static" salt water system below the fresh water lens should be approached with reservation. The stability depends upon many factors some of which are mentioned in the text (p. 3.5.18), some aren't. Examples of these factors are: amount of rainfall, frequency of rainfall, water usage (pumping regimes), tides, sea level fluctuations, and erosion.

In what sense is the ocean considered to provide an additional barrier?

4.d.6 p. 3.5.12

The statement that 85 percent of the world's earthquake energy is released in the Pacific margins should be documented.

4.d.7 p. 3.5.12

Figure 3.5.6 does not show major basement rock types. There is a figure showing major basement rock types in Reference 5 (Bayley and Muehlberger; 1968), which has Figure 3.5.6 as an inset, titled "Principle Basement Provinces."

4.d.8 p. 3.5.18

The discussion of sorptive phenomenon is not sufficiently covered. A comparison of the sorptive properties associated with island disposal with

Comment
Number

those associated with conventional geologic disposal should be presented, to determine if the multibarrier approach has been effectively utilized.

4.d.9 p. 3.5.19

It should be noted that dispersion and diffusion may be very active in this type of system, especially in combination with a natural zone of dispersion along the saltwater/freshwater interface.

4.d.10 p. 3.5.23

Under Section 3.5.2.2, some estimate should be provided of the probability of accidents on the sea lanes, which might lead to loss of the radioactive cargo. Cost estimates should also be provided.

4.d.11 p. 3.5.27

It should be noted that current models are not able to accurately predict flow through fractured media, which will be normally encountered in islands of volcanic origin.

4.d.12 p. 3.5.29

Section 3.5.6.3 identifies research and development areas that need to be explored in order to resolve uncertainties in island disposal. One area is the level of risk associated with extended sea transportation paths. Since the complexity of port facilities varies with the island disposal option being considered the level of risk, both in terms of routine occupational exposure and exposures due to accidents should also be considered as an area needing development.

4.d.13 p. 4.15

It is not accurate to state that the insular geologic surroundings are of inherently dynamic nature. This is not so especially for the east coast continental islands. East coast islands are probably less likely to contain, or be near, valuable resources than some of the west coast islands, thus lessening the possibilities of repository intrusion.

Comment
Number.d.14 p. 4.20

Table 4.5.2 presents preliminary estimates of the socioeconomic impact of the waste management options. An assumption stated under island disposal is that dockside shipping facilities will be constructed in a well established port area. For the no recycle option, packaged spent fuel will be shipped to the island disposal area. The recent NRC interim rule for safeguarding spent fuel shipments may prevent the use of well established port areas so that the conclusion reached, that the incremental impact is small, may not be valid.

4.e Sub-Seabed Geological Disposal Concept4.e.1 p. 3.6.1

It is stated that the goal "to aid in solving national and international legal and political problems" will be started only after the technical and environmental feasibility is demonstrated. Has this been factored into the schedule that has been developed for this program? What lead time and resources have been planned? Has the DOE participated in any international discussions of this problem. A description of the programs of other countries interested in seabed disposal would be helpful.

4.e.2 p. 3.6.2

The "difficulty of documenting a repository's location for future generations" is presented as a major disadvantage of the seabed concept. Explain why this would be any more difficult to do for seabed than for conventional geologic disposal?

4.e.3 p. 3.6.3

Two study areas were identified as having been chosen in the central North Pacific. Where are these study areas located? (Locate on a map.)

4.e.4 p. 3.6.3

The statement, "This region (the continental margin) is therefore unsuitable for consideration as a possible waste disposal site." is too

Comment
Number

final for such a large region and cannot be justified without detailed discussion.

A much more reasonable and specific statement is that made for fracture zones in the mid-ocean ridge: "On the basis of present knowledge, therefore, the fracture zones are not probable candidates as study sites."

Similarly the statement, "The abyssal plains...are therefore unacceptable for further consideration." should be modified.

4.e.5

p. 3.6.4

It is stated that: "Bottom currents in the MPG areas of the North Pacific are generally weak and variable." A reference should be provided. How weak and variable bottom currents affect emplacement, radionuclide migration, heat transfer, etc. should be discussed.

4.e.6

p. 3.6.4

The sediment thickness is reported to be 50 to 100 meters, while in Table 3.6.1 it is given as 100 to 300 meters.

4.e.7

p. 3.6.4

A statement is made regarding waste disposal in trenches: "...a plate being subducted would have moved only tens of kilometers during that time (250 to 500 thousand years) and would not be subducted fast enough for waste disposal purposes." This conclusion does not follow from the discussion proceeding it in the same paragraph.

- a. How far would the waste have to move during that time to be subducted fast enough for waste disposal purposes? Reference?
- b. What might the impact be of the waste not being subducted fast enough?

Comment
Number

4.e.8

Section 3.6.2.3

This section starts off with the identification of the barriers to the movement of radionuclides, then fails to discuss two of them: "any controlled modification of the medium," and "the benthic boundary layer." A discussion of these barriers should be provided.

4.e.9

p. 3.6.7

Previous reports on the U.S. seabed disposal program have not included the water column as a design barrier. Is it the program's intention to now identify the water column as a primary design barrier to radionuclide migration, or rather to investigate its properties as a barrier only for unexpected releases? In other words, do the conceptual plans allow for radionuclides to enter the water column during the period when they may present a hazard to man or the ecosystem? What is meant by inadvertent release? Scenarios leading to inadvertent release should be described.

4.e.10

p. 3.6.20

Under the discussion of the water column, it should be recognized that while the water column may not provide a barrier to migration, its enormous capability to dilute such releases below significant concentrations cannot be overlooked as a mitigative feature (See comment 4.e.9).

4.e.11

p. 3.6.21

The research and development costs to support the penetrometer emplacement concept are quoted as \$250 million, on page 3.6.21, and as \$60 million on page 3.6.31. The components of each figure should be given. What is the meaning of "state-of-the-art" (Figure 3.6.1) referring to penetrometer emplacement, given the quarter of a billion dollar research and development cost estimate?

4.e.12

p. 3.6.24

It should be made clear that tsunamis could pose no danger to a ship that was not in shallow, near shore waters, or near the source of tsunami. Even a large tsunami would probably not be noticed by a ship in mid-ocean

Comment
Number

because of the long wave length (typically hundreds of kilometers) and relatively small oceanic wave heights (usually less than a meter). A minor storm or just rough seas would pose greater danger in mid-ocean.

- 4.e.13 The basis for the following cost estimates should be provided (including the components and assumptions for each):
- a. "The resulting order-of-magnitude figure is \$200 million for the capital cost of handling 1800-3600 MTHM/hr" (p. 3.6.21).
 - b. The \$25 million/year operating cost (p. 3.6.21).
 - c. "It is estimated that the program can be completed in 25 years at an overall cost of about \$560 million including construction of one ship and a port facility" (p. 2.6.27). Details on the 25 year schedules should also be provided.
 - d. Each of the estimated costs of the multibarrier research and development program (Section 2.6.6.2).

- 4.e.14 Section 3.6.6.1
This section is labeled "Site Selection and Preparation" but nothing is mentioned of site preparation. What is involved in preparing a seabed site for use?

4.f The Ice Sheet Disposal Concept

- 4.f.1 p. 3.7.10
Under Section 3.7.1.5, the risks, hazards, and impacts of transporting HLW over ice in polar climates should be presented.

Comment
Number

4.g Reverse Well Disposal

- 4.g.1 p. 3.8.1
A brief paragraph on retrievability appears. There is no assurance that the liquid waste, once pumped into a porous medium, is totally retrievable. Invariably, a certain fraction of the waste will remain "captive" within the host rock. Total recovery, at any cost, is likely not attainable. A more detailed discussion focusing on the impact of partial recovery should appear.

- 4.g.2 p. 3.8.2
One suggested storage media is depleted hydrocarbon reservoirs. There are obvious problems with this, as additional hydrocarbon reservoirs are often found beneath depleted fields. Recovery from the underlying reservoirs would necessitate penetrating the liquid waste reservoir. As improved hydrocarbon recovery techniques are continually being developed, utilization of depleted hydrocarbon reservoir area storage medium may preclude recovery of otherwise-available natural resources.

Are there any other examples of porous fractured strata that could be used for deepwell injection that would give a more balanced treatment to this concept?

4.h Omitted Concept

- 4.h.1 p. 3.1.33
This section states "Thus, cost considerations dictate that the depth of emplacement should be minimized, whereas isolation requires that the depth be maximized." The first part of that statement is sufficiently clear. However, it is not clear that the second part of the statement is correct or if correct, significant. The support for this part of the statement is qualitative and intuitive rather than quantitative and rigorous.

Geological Survey Circular 779 states: "The suggestion of Winograd (1974) that waste be placed at relatively shallow depths (30 to several hundred

Comment
Number

meters) in the thick (as thick as 600 m) unsaturated zones of the arid Western United States deserves consideration." We concur.

The Teknekron, Inc. report prepared for PNL, "A Cost Optimization Study for Geologic Isolation of Radioactive Wastes," May 1979, does not indicate any significant advantages to great depths of burial except the reduced probability of repository disruption. If the large meteorite strike is truly improbable and if erosion and glaciation can be avoided (at least during the first 10's of thousands of years) then there may not be any advantages to great burial depths, only disadvantages.

The following questions should be addressed:

1. Are there regions of the U.S. otherwise suitable for a repository which can provide a safe environment for the waste at relatively shallow depths without a meaningful threat of interruption by natural events?
2. If so, what is the reduction of risk between such a repository and a deep repository (and what is the increase in cost)? What is the potential for an increase in confidence which could result in a more complete site characterization and simpler modeling of a shallow versus deep repository?
3. If not, what is the quantitative reduction in risk as a function of depth for a deep repository?

Comment
Number5. COMPARISON OF ALTERNATIVES5.1 General

Chapter 4 does not supply an adequate summary of the results of the first three chapters, much less a comparative assessment of the ten options in Chapter 3, and gives little if any guidance for judging the relative environmental and social impacts of the possible courses of action.

5.2 General

Analyses have been done for 1985 and 2000 when the first repository won't likely be operational until well after 1990. DOE/ET-0028 pg. 2.3, paragraph 4, states: "...these dates are not critical to waste management costs or environmental effects." This is probably true. However, they could have a significant effect on the comparison of conventional geologic disposal with other disposal options. This should be addressed in the GEIS.

5.3 General

The only alternative that is covered in any degree of detail is deep geologic disposal. While it is realized that less information is available for other alternatives, it appears they could be considered in more detail than these have been. For example, transportation impacts vary widely among alternatives yet generally are dismissed without much discussion as being insignificant. (See e.g., discussion for island and seabed on 3.6.24-3.6.25 and ice sheets on 3.7.10.)

5.4 General

As a document addressing various possible disposal media (i.e., siting options) on a generic basis, the GEIS does not provide the detailed discussions necessary to give the reviewer confidence in the conclusions drawn. Too much of what purports to be discussion of siting is in reality discussion of waste handling and processing. As an example of a GEIS with detailed discussions of siting options and impacts, see the Final Environmental Statement on Floating Nuclear Plants (NUREG-0056).

Comment
Number

- 5.5 p. 1.31
The names and qualifications of the people who comprised the "panel of experts" who were involved with the comparative assessment of the alternatives should be discussed.
- 5.6 pp. 1.31, F.6
The significance of the comparative analysis is clouded by the use of scales that are nonlinear with no relative scaling distributions given and nonindicative of acceptability (e.g., page 4.10 contains a statement that "... 'five' the maximum rating does not necessarily represent a 'good' situation...")
- 5.7 pp. 3.1.136, 3.3.3
Where there exist areas of uncertainty common to different alternatives they should be equally treated. For example on page 3.3.3 it states, "Information to satisfactorily assess the feasibility of the very deep hole concept is inadequate. This is not to say that the concept is not feasible, but there is not sufficient knowledge at present to confirm that radioactive waste can be isolated deep enough...to avoid transport of radioactive material to the biosphere. The main uncertainty is the lack of information about porosity, permeability and water conditions at great depths. "On page 3.3.1 of the GEIS it states that very deep hole disposal is considered flawed because more information is needed on groundwater systems, rock strength and sealing of holes over long periods of time. On the other hand it is argued on page 3.1.136 that "No long term significant impacts are expected to result from waste repositories described previously in this statement whether located in salt, granite, shale or basalt formation." It would appear the information needs stated for deep hole disposal would also exist for conventional geological disposal.

The technology for long-term sealing which has not been demonstrated for any of the three options, also does not receive uniform evaluation in the GEIS. For example, on page 3.3.28, of the GEIS it states: "Placement of an adequate plug within the hole does not constitute an adequate seal

Comment
Number

- because fracturing of the host formation during boring or shaft sinking may lead to a highly permeable annulus around the hole." Mined repositories and very deep holes share this problem. Hence, it should also be identified as a serious potential problem in the mined repository.
- 5.8 p. 3.1.246
In the last paragraph on page 3.1.246 it is stated that "Table 3.1.95 presents for conventional geological disposal the data used as a basis for scalar quantities in the comparative analysis discussion. Table 3.1.95 implies that there is "no data" in a number of key areas for making a comparative analysis. Based on this it would appear that (1) no substantive basis exists for making a rational comparison among disposal options and (2) there may not even be a sufficient basis for assessing the expected environmental impacts from conventional geological disposal.
- 5.9 p. 4.2
There seems to be a contradiction between the statement on page 4.2, second paragraph, which says: "Value judgments were required in at least two areas: 1) judgments relative to selection of the decision criteria and 2) judgments relative to selection of appropriate methods of measuring effects on criteria," and the statement in the footnote on page 4.2 which says: "Because these questions relate to the values of society and individuals they are avoided here where possible."
- 5.10 p. 4.4
Table 4.2.1 indicates that "nonhigh-level" TRU wastes cannot be disposed of by, among others, the very deep hole, island disposal, and subseabed disposal methods. It is not apparent why this is so. The GEIS should either present a rationale for requiring separate disposal methods or include "nonhigh-level" wastes in the wastes to be disposed of by those disposal methods. This is important because the current GEIS assumptions require that if disposal of HLW by the above methods is used, disposal in mined cavities in bedded salt also be an acceptable method.

Comment
Number

.11 p. 4.7
Beginning on page 4.7 eleven decision criteria are presented and discussed. One is called Ecosystem Impact and consist of two attributes. No rationale is given for selecting these particular measures as criteria. On p. 4.11, Table 4.5.1 states that available information on the physical and operating characteristics of the commercial waste management options is not sufficient to permit comparative assessment of these attributes. Appendix F does not give any primary production information. While Table 3.1.95 presents data used as a basis for scalar quantities in comparative analysis. They give a value of 5×10^{10} g dry organic matter for reversible ecological effects. There is no explanation of where this number comes from or why it is used except that on page 5.19 a formula is given for determining primary production.

5.12 Determining net primary production has no value in deciding which CWM option should be selected nor in making decisions at other levels in the CWM program, e.g., among geological substrates or particular sites within geological substrates.

5.13 "Years until operational" is picked as the major decision factor in selecting technology (page 1.36, 4.11). But, a basis for considering this to be an important factor, that is a near-term need, is not articulated. On page 5.1, it is indicated that alternatives have been ranked with respect to the ease and likelihood of implementation by "the design target date" to evaluate development status of technology. What this target date is is not revealed. This approach is backwards in any event as the GEIS should present information to support the determination of a need date or of need as a function of time and not evaluate options by assuming a need date.

5.14 p. 4.11
Table 4.5.1 indicates that insufficient data is available to compare ecosystem, aesthetic, and critical resource consumption impacts. These are among the most basic and fundamental, true environmental impacts. The

Comment
Number

majority of the remaining criteria are better described as policy considerations than as environmental factors, e.g., status of technology, cost of construction, policy and equity considerations. Thus, it appears that the final comparative analysis in this environmental impact statement drops out environmental factors and is based on the policy considerations. Environmental impacts, other than dose assessments, such as hydrologic impacts including water use and availability and impacts of construction and operation of the repository need more detailed discussion.

5.15 p. 4.44
There are references to: "some argue that public confidence would be lost..." and on the first paragraph, page 4.45: "some people argue that..." Are these people DOE staff, results of public survey, comment letters? Who "some people" are should be specified.

July 26, 1979

Dr. Colin A. Heath
Division of Waste Isolation
Mail Stop B-107
U.S. Department of Energy
Washington, D.C. 20545

Dear Dr. Heath:

Attached are HEW's comments on the Draft Environmental
Impact Statement on the Management of Commercially Gener-
ated Radioactive Waste (DOE/EIS-0046-D).

Please feel free to contact the respondent agencies
directly if you wish to discuss their comments further.

Sincerely yours,

Charles Custard
Director
Office of Environmental Affairs

OEA:CUSTARD:am:7/26/79

DEPARTMENT OF
HEALTH, EDUCATION, AND WELFARE
OFFICE OF THE SECRETARY
WASHINGTON, D. C. 20201

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DEPARTMENT OF HEALTH, EDUCATION, AND WELFARE
PUBLIC HEALTH SERVICE
FOOD AND DRUG ADMINISTRATION
ROCKVILLE, MARYLAND 20857

July 3, 1979

Dr. Vernon Houk
Principal Environmental Officer/H
Center for Disease Control
Atlanta, Georgia 30333

Dear Dr. Houk:

The following are comments on the radiation aspects of the Draft Environmental Impact Statement (DOE/EIS-0046-D) for Management of Commercially Generated Radioactive Waste, dated April 1979.

1. In contrast to the impact statement for the WIPP project (DOE/EIS-0026-D) DOE in this DEIS has discussed applicable existing standards (section 2.2) and in Appendix G has described the relationship of existing standards. The discussions cited cover the current radiation protection practices but have not clearly stated the criteria that DOE would apply to operation of the high-level waste repository for routine operation, accidents, transportation of high-level waste, the sealing of the repository and the post-implacement period. Further, no mention is made of the role of the Environmental Protection Agency in setting general environmental protection standards applicable to the management of high-level wastes.

2. Models used in the DEIS are described in Appendix D, Appendix E, and Appendix I. Since the estimation of population exposure from specific radionuclides and from environmental pathways are based on computer codes, a statement should be made on the uncertainty of the data bases. Buried in the DEIS on page 3.1.137 is a statement that because of model uncertainties the range of accuracy of doses is estimated to be such that a given dose result for a regional population may be underestimated as much as a factor of 10 and may be overestimated as much as a factor of 100. In order to help the reader of the DEIS understand the uncertainties these qualifying factors should be included in Appendices D, E, and I. Also as footnotes to each Table that presents regional population doses. For example, Tables 3.1.84 to 3.1.87.

The environmental analysis of the total systems involved in waste disposal in geologic repositories is described in Section 3.1.5. In assessing the public health and radiation aspects of the various options under consideration it is evident from the DEIS that there are many tables and statements in the text that present 70-year dose estimates for the various waste management options and accident situations. Each table presented represents the radiation impact for the particular consideration

Dr. Vernon Houk

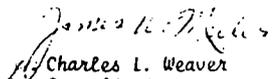
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being analyzed. The radiation safety of waste repository operations could be critical consideration in the selection of a waste management option. It would be helpful if DOE could provide summary tables that would show the range of environmental impacts for the conventional geologic disposal presented in section 3.1. For example, it is possible to identify the range of radiation dose and health effects from routine operation from repositories in salt, granite, shale and basalt using Tables 3.1.84 - 3.1.87. For repository startup in 1985 the regional population doses (70-year total body) ranges from 3×10^3 to 6×10^4 man-rem and health effects range from 6 to 50 (no health effects are shown for the once-through option). A similar range for the repository start-up in the year 2000 shows that the regional population dose ranges from 3×10^3 to 2×10^4 man-rem and that the health effects range from 6 to 50. From the above, there is only a factor of 20 difference in regional population dose for the various options for year 1985 start-up and a factor of 6 difference for year 2000 start-up.

Radiation dose summaries in a separate section of the DEIS could provide a better understanding of the health impact from the various waste management options. Furthermore, in view of the public awareness of radiation dose and health effects as a result of the Three Mile Island accident this aspect of the DEIS should be reexamined to assure that the radiation and public health aspects have been adequately assessed and presented in a form that the public would understand.

We have no specific comments on the alternate for final disposal described in sections 3.2 through 3.10. For assessment of long-term radiological safety sufficient data were not available to use the hazard index methodology for any of the commercial waste management (CWM) options according to the statement in section 4.5.3 (page 4.14). However, the comparative analysis in section 4.5 as summarized in Table 4.5.1 states that the comparative assessments were based on estimates by a panel of experts.

Sincerely yours,


Charles L. Weaver
Consultant
Bureau of Radiological Health

cc: Dr. Kenneth Taylor, HFV-2