DOE/EIS-0220

Final Environmental Impact Statement

Interim Management of Nuclear Materials

October 1995

Department of Energy
Savannah River Site
Aiken, South Carolina

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COVER SHEET

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ABSTRACT: This document evaluates the potential environmental impacts of alternatives for the stabilization of nuclear materials currently stored at various locations on the Savannah River Site (SRS). These materials remain from past defense-related production, testing, and other activities at the SRS and from chemical separations and related activities that DOE suspended in 1992. The EIS analyzes the following alternatives: Continuing Storage (No Action), Processing to Metal, Processing to Oxide, Blending Down to Low Enriched Uranium, Processing and Storage for Vitrification in the Defense Waste Processing Facility, Vitrification (F-Canyon), and Improving Storage. The preferred alternatives cover a combination of these in relation to the different types of material. TC

PUBLIC COMMENTS: In its preparation of this Final Environmental Impact Statement, DOE considered written comments sent to DOE, comments submitted by voice mail, and comments received at public meetings held in Savannah, Georgia (April 11, 1995) and North Augusta, South Carolina (April 13, 1995). TC

FOREWORD

Note: Revisions are indicated in the Final EIS by vertical change bars in the margin, which are cross-referenced to the specific comments presented in Appendix F. Appendix F also contains the DOE responses to comments and cross-references to appropriate sections of the EIS. Other change bars are marked either TC (technical change) or TE (editorial change). TC

The Savannah River Site (SRS) is a major Department of Energy (DOE) installation. The past mission of the SRS was to produce nuclear materials that supported the defense, research, and medical programs of the United States.

In 1992 the Secretary of Energy directed the SRS to phase out defense-related chemical separations activities. As a result of shutdowns and reduced demand for nuclear materials, the SRS presently has a large inventory of in-process radioactive solutions, nuclear reactor fuel assemblies, and reactor targets. These materials, due to their form or to the condition in which they are maintained, could represent a concern for the public, worker health and safety, and the environment. TE

DOE published a Notice of Intent (NOI) to prepare this environmental impact statement (EIS) on March 17, 1994 (59 FR 12588). The purposes of DOE actions related to the inventory of nuclear materials at the SRS are to stabilize those materials that represent a health and safety concern for the public, workers, and the environment in the short term and to convert those materials required to support DOE programs to the desired products. On June 21, 1994, DOE issued an NOI to prepare a *Programmatic Environmental Impact Statement for Storage and Disposition of Weapons-Usable Fissile Materials* (59 FR 31985). DOE anticipates that it will need as long as 10 years to begin the implementation of the decisions it makes as a result of that programmatic EIS. In the meantime, some of the materials at the SRS require continuing vigilance because of unstable configurations and uncertainties related to continued storage. DOE needs to take the intermediate steps evaluated in this site-specific EIS before it can make and implement long-term decisions on the management and disposition of nuclear materials. TE

The NOI for this EIS requested public comments and suggestions for DOE to consider in its determination of the scope of the EIS, and announced a public scoping period that ended on May 31, 1994. During the scoping period, individuals, organizations, and government agencies submitted 80 comments that DOE considered applicable to the interim management of nuclear materials. In addition, DOE held scoping meetings in Savannah, Georgia; North Augusta, South Carolina; and Columbia, South Carolina, on May 12, 17, and 19, 1994, respectively. TC

In response to the comments received during the scoping process, DOE evaluated the environmental impacts that could occur from alternatives for the management of the nuclear materials at the SRS that would be initiated over the next 10 years. DOE used the 10-year period because the public felt it might take that long for DOE to begin the implementation of long-term management and disposition plans.

TC

On March 17, 1995, the U.S. Environmental Protection Agency published a Notice of Availability (NOA) in the *Federal Register* (60 FR 52, pages 14432-14433), which officially started the public comment period on the Draft Interim Management of Nuclear Materials EIS; DOE published a corresponding NOA for the Draft EIS on April 6, 1995 (60 FR 66, pages 17523-17524). The public comment period ended on May 1, 1995.

DOE has prepared this EIS in accordance with the NEPA regulations of the Council on Environmental Quality (40 CFR Parts 1500-1508) and DOE NEPA Implementing Procedures (10 CFR Part 1021). This EIS identifies the methods used and the scientific and other sources of information consulted. In addition, it incorporates, physically or by reference, available results of ongoing studies. DOE has revised the Draft EIS as appropriate in response to comments received electronically, in letters, and during public hearings in Savannah, Georgia, and North Augusta, South Carolina (April 11 and 13, 1995, respectively), from individuals, organizations, and Federal and state agencies.

The organization of the EIS is as follows:

- The "Summary" is a general overview of the information in the EIS. It contains tables from Chapter 2 that summarize the total estimated impacts of ongoing management actions or alternatives that would be implemented over the next 10 years from normal operations for each category (or type) of nuclear material. These tables also list estimated impacts (latent cancer fatalities) from the potential accident with the highest consequences associated with each alternative. TC

- Chapter 1, "Purpose and Need for Action," describes the purpose and need for interim nuclear material management activities. This chapter also identifies and categorizes the nuclear materials addressed in this EIS.

- Chapter 2, "Alternatives," identifies the alternatives that DOE could use for the management of the nuclear materials at the SRS. As indicated above, Chapter 2 contains tables that list the total estimated impacts from normal operations that could be implemented over the next 10 years for each category (or type) of nuclear material. These tables also list estimated impacts (latent cancer fatalities) from the potential accident with the highest consequences associated with each alternative.

- Chapter 3, "Affected Environment," describes the SRS environment as it relates to the alternatives discussed in Chapter 2.

- Chapter 4, "Environmental Impacts," assesses the potential environmental impacts of the alternatives under normal operation and accident conditions. DOE has reformatted Chapter 4 to enhance the clarity of presentation. In the Draft EIS, Chapter 4 presented a range of impacts by scenario (No Action, Preferred, or Comparative). Each scenario (with the exception of No Action) identified a stabilization action for each material of interest. Because these scenarios really represent "cumulative" impacts for various combinations of alternatives, DOE has moved them from Chapter 4 to Chapter 5. As in the Summary and Chapter 2, Chapter 4 also contains tables of estimated impacts, but these tables include additional parameters. (For example, the tables in the Summary and Chapter 2 list the incremental contribution each alternative could make to the highest annual concentration of nitrogen oxide in the air, measured at the SRS boundary. The Chapter 4 tables list the highest 1-hour, 3-hour, 8-hour, 12-hour, 24-hour, weekly, or annual average concentration of the pollutants of concern in the air around the SRS. The tables in the Summary and Chapter 2 list only nitrogen oxide because it is typically the nonradiological air pollutant of primary interest. In other words, the impact data shown in the Summary and Chapter 2 tables are a subset of the more detailed impact data presented in Chapter 4.) For alternatives that would be initiated but not completed within the 10-year period, DOE has provided information on the long-term impacts of the alternative under normal operating conditions. For example, alternatives involving vitrification in the Defense Waste Processing Facility would not be completed for many years due to the existing volume of high-level waste requiring vitrification. However, DOE has estimated the number of high-level waste canisters resulting from this alternative that would require placement in a geologic repository. TC

- Chapter 5, "Cumulative Impacts," discusses the cumulative impacts of interim management actions in relation to impacts of past, present, and foreseeable future activities at the SRS. As described above, DOE has reformatted Chapter 5 to enhance the clarity of presentation by incorporating the scenarios originally presented in Chapter 4. In addition, as the result of stakeholder input DOE has added a new scenario, Minimum Processing, to this discussion.

- Chapter 6, "Short-Term Use Versus Long-Term Resource Commitments," presents the commitments associated with reinstituting activities in the F- and H-Canyons and support facilities.

- Chapter 7, "Irreversible or Irretrievable Resource Commitments," identifies such commitments associated with the interim management of nuclear materials.

- Chapter 8, "Applicable Laws, Regulations, and Other Requirements," discusses regulatory requirements, including applicable statutes and DOE Orders, and compliance with state and Federal regulations.

- Appendix A, "Radioactive Materials at the Savannah River Site," discusses radiological vulnerabilities associated with some of the materials, and identifies SRS nuclear materials in three categories: (1) Stable (material that DOE can store safely as it currently exists), (2) Programmatic (material for which DOE has identified a potential use), and (3) Candidates for Stabilization (material in a storage condition that DOE has identified as an environmental, safety, and health concern). Appendix A also discusses the quantities of these materials at the SRS. TC

- Appendix B, "Radioactivity and Radiological Health Effects," contains, in response to a public comment, a basic discussion of radiation and its health effects. DOE has deleted the original Appendix B, "Programmatic Need for and Use of Plutonium-242," which appeared in the Draft EIS. DOE has decided to deal with such long-term programmatic issues in an upcoming document, the Stockpile Stewardship and Management Programmatic Environmental Impact Statement (discussed in Chapter 1).

- Appendix C, "Facility and Process Descriptions," describes facilities and processes that would be involved in the interim management of nuclear materials.

- Appendix D, "Annual Data for Phases Associated with the Management of Materials," provides environmental impact data for normal operations related to the interim management of nuclear materials. Appendix D contains annual estimates for the phases of each alternative. DOE used the data in Appendix D, along with representative duration of potential activities, to estimate the 10-year impacts for each alternative.

- Appendix E, "Accidents," discusses accidents that could occur at SRS facilities during the interim management of nuclear materials. Similar to Appendix D, Appendix E analyzes environmental impacts from potential accidents associated with the phases of each alternative. The tables in Appendix E use **bold** type to identify the accidents that pose either the greatest risk (considering both the likelihood and the consequences) or the greatest consequence (in latent cancer fatalities). DOE included the highest consequence accident for each alternative in the Summary and Chapter 2 tables.

- Appendix F, "Public Comments and Responses," is a new appendix that presents the text of public comments received on the Draft EIS and the DOE responses to these comments.

Some changes to the Draft EIS affect several chapters of the Final EIS. For example, DOE has reconsidered its designation of the preferred alternatives for the Mark-16 and -22 fuel and the other aluminum-clad targets in the Draft EIS. In part in response to public comments, DOE determined that it needs further analysis of the relative costs of implementing the interim management alternatives evaluated in this EIS, and the optimum of the alternatives. Thus DOE is conducting further analyses regarding costs and timing for interim management alternatives for these materials. DOE does not anticipate the development of any new information during this process that would affect the analyses of potential environmental impacts from managing the Mark-16 and -22 fuels and other aluminum-clad targets. Nevertheless, until the cost and schedule information is available to the decisionmaker, DOE's preferred alternative for these materials is Continuing Storage (No Action). DOE expects the cost and schedule information to be available in approximately 2 to 4 months. Before making a decision on the stabilization of the Mark-16 and -22 fuels and other aluminum-clad targets, DOE will announce its preferred alternative and allow a 30-day waiting period before issuing its Record of Decision.

TC

In addition, the Draft EIS stated that all material in the Receiving Basin for Offsite Fuel (RBOF) was suitable for extended storage for the next 10 years. In response to public comments, DOE has reevaluated its designation of some of the material stored in the RBOF. As a result of this reevaluation, DOE has identified a small amount of fuel and target material as a candidate for stabilization. This material consists of 81 canisters of failed Taiwan Research Reactor fuel, a failed canister of Experimental Breeder Reactor-II slugs, and six Mark-31 target slugs.

Transcripts of public testimony, copies of letters, comments and DOE responses, and reference materials cited in this EIS are available for review in the DOE Public Reading Room at the University of South Carolina-Aiken Campus, Gregg-Graniteville Library, 2nd Floor, University Parkway, Aiken, South Carolina, (803) 648-6851, and at the Freedom of Information Reading Room, Room 1E-190, Forrestal Building, 1000 Independence Avenue, S.W., Washington, D.C., (202) 586-6020.

TABLE OF CONTENTS

Chapter Page

COVER SHEET iii

FOREWORD v

SUMMARY S-1

1. PURPOSE AND NEED FOR ACTION 1-1

1.1 Introduction and Background 1-1
1.2 Purpose and Need for Action 1-8
1.3 Defense Nuclear Facilities Safety Board Review 1-9
1.4 Categories of Nuclear Materials 1-11
1.5 Categorization Methods 1-13

1.5.1 Stable Materials and Candidates for Stabilization 1-13
1.5.2 Programmatic Materials 1-22

1.6 Related National Environmental Policy Act Documents 1-25
1.7 Relationship of Decisions 1-32

2. ALTERNATIVES 2-1

2.1 Stable Material 2-4
2.2 Programmatic Material 2-5

2.2.1 Plutonium-242 2-6
2.2.2 Americium and Curium 2-11
2.2.3 Neptunium-237 2-15

2.3 Candidate Materials for Stabilization 2-17

2.3.1 H-Canyon Plutonium-239 Solutions 2-18
2.3.2 H-Canyon Uranium Solutions 2-25
2.3.3 Plutonium and Uranium Stored in Vaults 2-27
2.3.4 Mark-31 Targets 2-31
2.3.5 Mark-16 and -22 Fuels 2-37
2.3.6 Other Aluminum-Clad Targets 2-40
2.3.7 Aluminum-Clad Taiwan Research Reactor Fuel and Experimental Breeder Reactor-II Slugs 2-41

2.4 Comparison of Environmental Impacts Among the Alternatives 2-47

2.4.1 Health Effects 2-47
2.4.2 Air and Water Resources 2-59
2.4.3 Utilities 2-59
2.4.4 Waste 2-59
2.4.5 Contributors to Environmental Impacts 2-61

TABLE OF CONTENTS (continued)

Chapter Page

2.5 Other Factors 2-63

2.5.1 New Facilities Required 2-64
2.5.2 Security and Nonproliferation 2-65
2.5.3 Implementation Schedule 2-67
2.5.4 Technology Availability and Technical Feasibility 2-80
2.5.5 Labor Availability and Core Competency 2-80
2.5.6 Aging Facilities 2-81
2.5.7 Minimum Custodial Care 2-83

2.6 Cost 2-83
2.7 Preferred Alternatives 2-90

2.7.1 Stable Material 2-92
2.7.2 Plutonium-242 2-92
2.7.3 Americium and Curium 2-93
2.7.4 Neptunium-237 2-93
2.7.5 H-Canyon Plutonium-239 Solutions 2-94
2.7.6 H-Canyon Uranium Solutions 2-95
2.7.7 Plutonium and Uranium Stored in Vaults 2-96
2.7.8 Mark-31 Targets 2-97
2.7.9 Mark-16 and -22 Fuel 2-99
2.7.10 Other Aluminum-Clad Targets 2-101
2.7.11 Taiwan Research Reactor Fuel and Experimental Breeder Reactor-II Slugs 2-102

2.8 Other Alternatives Considered 2-103

2.8.1 Interim Canning and Storage of Fuel and Targets 2-104
2.8.2 Potential Processing Technologies 2-105

2.9 Facility Utilization 2-106

3. AFFECTED ENVIRONMENT 3-1

3.1 Geologic Setting and Seismicity 3-4

3.1.1 Subsurface Features 3-4
3.1.2 Seismicity 3-7

3.2 Water Resources 3-8

3.2.1 Surface-Water and Groundwater Features 3-8
3.2.2 Surface-Water and Groundwater Quality 3-10

3.3 Air Resources 3-11

3.3.1 Severe Weather Conditions 3-13
3.3.2 Radiological Air Quality 3-13
3.3.3 Nonradiological Air Quality 3-14

TABLE OF CONTENTS (continued)

Chapter Page

3.4 Socioeconomics 3-18

3.4.1 Employment 3-18
3.4.2 Population 3-18
3.4.3 Community Characteristics 3-19

3.5 Public and Worker Health 3-22

3.5.1 Public Radiological Health 3-22
3.5.2 Public Nonradiological Health 3-25
3.5.3 Worker Radiological Health 3-26
3.5.4 Worker Nonradiological Health and Safety 3-27

4. ENVIRONMENTAL IMPACTS 4-1

4.1 Impacts from Normal Operations 4-2

4.1.1 Health Effects from Normal Operations 4-17
4.1.2 Air Resources 4-46
4.1.3 Water Resources 4-52
4.1.4 Utilities 4-54
4.1.5 Waste Management 4-56
4.1.6 Traffic 4-63

4.2 Impacts from Accidents 4-63

4.2.1 Impacts from Facility Accidents 4-63
4.2.2 Radiological Impacts from Transportation Accidents 4-71
4.2.3 Environmental Justice Accident Assessment 4-77

4.3 Impacts Associated with Construction of a Dry Storage Facility 4-77

4.3.1 Land Use 4-78
4.3.2 Socioeconomics 4-78
4.3.3 Cultural Resources 4-78
4.3.4 Aesthetics and Scenic Resources 4-79
4.3.5 Geology 4-79
4.3.6 Air Quality 4-79
4.3.7 Water Resources 4-80
4.3.8 Ecology 4-81
4.3.9 Noise 4-81
4.3.10 Traffic and Transportation 4-82
4.3.11 Utilities and Energy 4-82

5. CUMULATIVE IMPACTS 5-1

5.1 Cumulative Impacts of DOE Actions Associated with the Interim Management of Nuclear Materials 5-1

5.1.1 Public and Worker Health 5-5
5.1.2 Air Resources 5-6
5.1.3 Water Resources 5-8

TABLE OF CONTENTS (continued)

Chapter Page

5.1.4 Waste Generation 5-8
5.1.5 Utilities and Energy 5-9

5.2 Cumulative Impacts of Other Reasonably Foreseeable Actions 5-10

5.2.1 Public and Worker Health 5-12
5.2.2 Air Resources 5-13
5.2.3 Water Resources 5-15
5.2.4 Waste Generation 5-16
5.2.5 Utilities and Energy 5-17

6. SHORT-TERM USE VERSUS LONG-TERM RESOURCE COMMITMENTS 6-1

7. IRREVERSIBLE OR IRRETRIEVABLE RESOURCE COMMITMENTS 7-1

7.1 Materials 7-1
7.2 Energy 7-1

8. APPLICABLE LAWS, REGULATIONS, AND OTHER REQUIREMENTS 8-1

8.1 Laws and Other Requirements 8-1

8.1.1 Federal Environmental Statutes and Regulations 8-1
8.1.2 Executive Orders 8-8
8.1.3 DOE Regulations and Orders 8-9

8.2 Emergency Management and Response 8-10

8.2.1 Authorities and Directives 8-10
8.2.2 Executive Orders 8-14
8.2.3 Emergency Planning Documents 8-15

Appendix A Radioactive Materials at the Savannah River Site

Appendix B Radioactivity and Radiological Health Effects

Appendix C Facility and Process Descriptions

Appendix D Annual Data for Phases Associated with the Management of Materials

Appendix E Accidents

Appendix F Public Comments and Responses

References

List of Preparers and Contributors

TABLE OF CONTENTS (continued)

Glossary

Acronyms and Abbreviations

Distribution List

Index

LIST OF TABLES

Table Page

1-1 SRS nuclear materials 1-24
2-1 Alternatives for the management of SRS nuclear materials 2-2
2-2 Comparison of the potential environmental impacts of the alternatives for stable material 2-48
2-3 Comparison of the potential environmental impacts of the alternatives for plutonium-242 2-49
2-4 Comparison of the potential environmental impacts of the alternatives for americium and curium 2-50
2-5 Comparison of the potential environmental impacts of the alternatives for neptunium 2-51
2-6 Comparison of the potential environmental impacts of the alternatives for H-Canyon plutonium-239 solutions 2-52
2-7 Comparison of the potential environmental impacts of the alternatives for H-Canyon enriched uranium solutions 2-53
2-8 Comparison of the potential environmental impacts of the alternatives for plutonium and uranium stored in vaults 2-54
2-9 Comparison of the potential environmental impacts of the alternatives for Mark-31 targets 2-55
2-10 Comparison of the potential environmental impacts of the alternatives for Mark-16 and -22 fuels 2-56
2-11 Comparison of the potential environmental impacts of the alternatives for other aluminum-clad targets 2-57
2-12 Comparison of the potential environmental impacts of the alternatives for Taiwan Research Reactor fuel and Experimental Breeder Reactor-II slugs 2-58
2-13 Comparison of environmental factors between separations facilities in F- and H-Areas under operating conditions 2-62
2-14 Composition of management scenarios 2-68
2-15 Estimated costs to manage generated wastes 2-90
3-1 1993 liquid releases by source (including direct and seepage basin migration releases) 3-11
3-2 Radioactivity in air at the SRS boundary and at the 160-kilometer (100-mile) radius during 1993 3-14
3-3 Average atmospheric tritium concentrations around the Savannah River Site 3-14
3-4 1993 atmospheric releases by operational group 3-15
3-5 Estimated ambient concentrations of criteria air pollutants from SRS sources 3-17
3-6 Estimated 24-hour average ambient concentrations at the SRS boundary - toxic air pollutants regulated by South Carolina from SRS sources 3-17
3-7 General racial characteristics of population in the SRS region of interest 3-20
3-8 General poverty characteristics of population in the SRS region of interest 3-20
3-9 SRS annual individual and collective radiation doses 3-27
3-10 Estimated maximum annual concentrations of workplace pollutants regulated by the Occupational Safety and Health Administration 3-28
4-1 Data (10-year totals) for impact analysis of various stabilization alternatives for stable material (from Table D-2) 4-4
4-2 Data (10-year totals) for impact analysis of various stabilization alternatives for plutonium-242 (from Tables D-3 through D-7) 4-5
4-3 Data (10-year totals) for impact analysis of various stabilization alternatives for americium and curium (from Tables D-8 through D-14) 4-6

LIST OF TABLES (continued)

Table Page

4-4 Data (10-year totals) for impact analysis of various stabilization alternatives for neptunium (from Tables D-15 through D-18) 4-7
4-5 Data (10-year totals) for impact analysis of various stabilization alternatives for H-Canyon plutonium-239 solutions (from Tables D-19 through D-23) 4-8
4-6 Data (10-year totals) for impact analysis of various stabilization alternatives for H-Canyon enriched uranium solutions (from Tables D-24 through D-27) 4-9
4-7 Data (10-year totals) for impact analysis of various stabilization alternatives for plutonium and uranium stored in vaults (from Tables D-28 through D-37) 4-10
4-8 Data (10-year totals) for impact analysis of various stabilization alternatives for Mark-31 targets (from Tables D-38 through D-43) 4-11
4-9 Data (10-year totals) for impact analysis of various stabilization alternatives for Mark-16 and -22 fuels (from Tables D-44 through D-48) 4-12
4-10 Data (10-year totals) for impact analysis of various stabilization alternatives for other aluminum-clad targets (from Tables D-49 through D-51) 4-13
4-11 Data (10-year totals) for impact analysis of various stabilization alternatives for Taiwan Research Reactor fuel and Experimental Breeder Reactor-II slugs (from Tables D-55 through D-61) 4-14
4-12 Impact assessment methodology for normal operations 4-15
4-13 Comparison of potential 10-year radiological health effects of the alternatives for stable material (from Table 4-1) 4-19
4-14 Comparison of potential 10-year radiological health effects of the alternatives for plutonium-242 (from Table 4-2) 4-20
4-15 Comparison of potential 10-year radiological health effects of the alternatives for americium and curium (from Table 4-3) 4-21
4-16 Comparison of potential 10-year radiological health effects of the alternatives for neptunium (from Table 4-4) 4-22
4-17 Comparison of potential 10-year radiological health effects of the alternatives for H-Canyon plutonium-239 solutions (from Table 4-5) 4-23
4-18 Comparison of potential 10-year radiological health effects of the alternatives for H-Canyon enriched uranium solution (from Table 4-6) 4-24
4-19 Comparison of potential 10-year radiological health effects of the alternatives for plutonium and uranium stored in vaults (from Table 4-7) 4-25
4-20 Comparison of potential 10-year radiological health effects of the alternatives for Mark-31 targets (from Table 4-8) 4-26
4-21 Comparison of potential 10-year radiological health effects of the alternatives for Mark-16 and -22 fuels (from Table 4-9) 4-27
4-22 Comparison of potential 10-year radiological health effects of the alternatives for other aluminum-clad targets (from Table 4-10) 4-28
4-23 Comparison of potential 10-year radiological health effects of the alternatives for Taiwan Research Reactor and Experimental Breeder Reactor-II fuel (from Table 4-10) 4-29
4-24 Permissible Exposure Limits (milligrams per cubic meter) of nonradiological air pollutants regulated by the Occupational Safety and Health Administration 4-34
4-25 Latent cancer fatalities resulting from routine shipment of nuclear materials 4-35
4-26 Latent cancer fatalities resulting from routine shipment of radioactive waste 4-37
4-27 Annular sector factors for local dose evaluations 4-44
4-28 Estimated per capita 10-year dose for identified communities in 80-kilometer (50-mile) region 4-45

LIST OF TABLES (continued)

Table		Page
4-29	Regulatory standards for nonradiological air pollutants (micrograms per cubic meter)	4-51
4-30	Estimated impacts from maximum projected consequence accidents (latent cancer fatalities) for each alternative	4-66
4-31	Estimated impacts from maximum projected risk accidents (latent cancer fatalities per year) for each alternative	4-68
4-32	Estimated latent cancer fatalities from medium probability accidents involving shipment of nuclear material	4-72
4-33	Estimated latent cancer fatalities from medium severity accidents involving shipment of radioactive waste	4-74
4-34	Air pollutant concentrations (micrograms per cubic meters) due to construction activities	4-80
5-1	Composition of management scenarios	5-2
5-2	Data for impact analysis of potential scenarios of alternatives	5-3
5-3	Environmental data associated with construction of the Dry Storage Facility	5-4
5-4	Estimated 10-year radiological health effects from normal operations	5-5
5-5	Estimated radiological doses from airborne releases of radioactivity associated with each management scenario	5-7
5-6	Estimated radiological doses from surface-water pathway exposures	5-8
5-7	Estimated average annual waste generated over the 10-year time period by scenario	5-9
5-8	Current and estimated utilities and energy usage associated for management scenarios	5-10
5-9	Estimated average annual cumulative radiological doses and resulting health effects to offsite population and facility workers	5-13
5-10	Estimated maximum nonradiological cumulative ground-level concentrations of criteria and toxic pollutants (micrograms per cubic meter) at the SRS boundary	5-14
5-11	Estimated average annual cumulative radiological doses and resulting health effects to offsite population from airborne releases	5-14
5-12	Estimated average annual cumulative radiological doses and resulting health effects to offsite population from liquid releases	5-15
5-13	Estimated cumulative waste generation from SRS operations	5-16
5-14	Estimated average annual cumulative utility consumption	5-17
8-1	DOE Orders Relevant to the Interim Management of Nuclear Materials	8-11

LIST OF FIGURES

Figure Page

1-1 Location of the Savannah River Site 1-2
1-2 Locations of principal facilities at the Savannah River Site 1-3
1-3 Historic nuclear materials production cycle at the Savannah River Site 1-4
1-4 Historic scrap recovery cycle for plutonium at the Savannah River Site 1-6
1-5 Historic reprocessing cycle for offsite spent fuel at the Savannah River Site 1-7
1-6 Public risk profile before and after stabilization actions 1-9
1-7 Amount of nuclear material in each category 1-25
2-1 H-Area 2-7
2-2 F-Area 2-12
2-3 Integrated Schedule - No-Action Scenario 2-70
2-4 Integrated Schedule - Preferred Scenario 2-72
2-5 Integrated Schedule - Minimum Processing Scenario 2-75
2-6 Integrated Schedule - Comparative Scenario 2-78
2-7 10-Year Cost Profile - No-Action Scenario 2-84
2-8 10-Year Cost Profile - Minimum Processing Scenario 2-85
2-9 10-Year Cost Profile - Preferred Scenario 2-86
2-10 10-Year Cost Profile - Comparative Scenario 2-87
3-1 Representative site for Dry Storage Facility 3-2
3-2 Savannah River Site, showing principal industrial areas, roads, and railroads 3-3
3-3 General location of the Savannah River Site and relationship to physiographic provinces of the eastern United States 3-5
3-4 Savannah River Site, showing seismic fault lines and locations of onsite earthquakes 3-6
3-5 Savannah River Site, showing 100-year floodplain and major stream systems 3-9
3-6 Groundwater contamination at the Savannah River Site 3-12
3-7 Distribution of people of color by census tracts in the SRS region of analysis 3-21
3-8 Low-income census tracts in the SRS region of analysis 3-23
3-9 Major sources of radiation exposure in the vicinity of the Savannah River Site 3-24
4-1 Annular sectors around the Savannah River Site 4-42
4-2 Community impacts from a unit population dose 4-45

SUMMARY

The Atomic Energy Commission (AEC), a predecessor agency of the Department of Energy (DOE), established the Savannah River Site (SRS) in the early 1950s to produce special radioactive isotopes. In this regard, the primary SRS mission was the production of strategic isotopes (plutonium-239 and tritium) used in the development and production of nuclear weapons for national defense. The Site produced other special isotopes (californium-252, plutonium-238, americium-241, etc.) to support research in nuclear medicine, space exploration, and commercial applications. The historic production cycle at the SRS involved the fabrication of metal fuel and target assemblies for irradiation in the Site's nuclear reactors, followed by chemical dissolution, separation, and conversion of the radioisotopes into solid forms for use at the SRS or other DOE sites. TE TE

In March 1992, DOE suspended chemical separations activities at the SRS to address a potential safety concern regarding the capacity of the ventilation systems in F- and H-Canyons to withstand an earthquake. That concern was addressed; however, before the resumption of reprocessing, the Secretary of Energy directed that the SRS phase-out defense-related chemical separations activities in those facilities (DOE 1992a). World events during the late 1980s and early 1990s resulted in the end of the Cold War and a reduction in the demand for new materials for nuclear weapons. Thus, after March 1992 DOE did not process nuclear materials at the SRS chemical separations facilities to recover special isotopes, with the exception of scrap materials containing plutonium-238. DOE continued these plutonium-238 operations to support future National Aeronautics and Space Administration (NASA) exploratory space missions. In February 1995, DOE resumed F-Canyon operations to eliminate the risks involved in storing plutonium in liquid solutions (DOE 1994d). TC

The cessation of processing operations resulted in a large inventory of nuclear materials being caught in various stages of the historic SRS production cycle (fabrication, irradiation, reprocessing, and recovery). These materials include irradiated and unirradiated reactor fuel, targets, and components; solutions containing dissolved nuclear materials and recovered isotopes in stainless-steel tanks; and product and scrap forms of metals or oxides in containers (cans, drums, etc.) typically used for temporary storage or shipment off the Site.

Differences Between Draft EIS and Final EIS

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DOE published the Draft Interim Management of Nuclear Materials Environmental Impact Statement in March 1995. The Environmental Protection Agency (EPA) announced the availability of the Draft EIS for public review and comment in the *Federal Register* on March 17, 1995 (60 FR 52, pages 14432-14433); this announcement initiated a 45-day public comment period on the Draft EIS that ended on May 1, 1995.

DOE held a series of meetings to receive comments and to exchange information with the public on the Draft EIS. In addition to the comments received at the public meetings, DOE also received comments from government agencies, private organizations, and individuals. DOE has included all the comments it received in Appendix F along with the Department's responses. The comments generally pertain to the following:

- The materials included in the EIS and their categorization (i.e., stable, programmatic, or candidates for stabilization)

- The choice of a preferred alternative for a given material group and the method used to select a preferred alternative

- Additional alternatives for some material groups

- Costs of the alternatives for managing the nuclear materials at the SRS

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- The inclusion of additional technical information, such as clarification of analysis techniques of health and environmental impacts; further discussion of alternatives; consideration of another scenario; and discussion of safety issues regarding SRS facilities and nuclear materials

- Questions on the selection of 10 years as the length of time DOE would require to make and implement decisions

In response to these comments, DOE made changes to the EIS, as summarized below:

- Chapter 2 includes additional information on the materials included in the EIS and how DOE categorized the materials. DOE recategorized two groups of materials. The first group consists of a limited amount of Taiwan Research Reactor (TRR) fuel and Experimental Breeder Reactor (EBR) slugs that could be leaking radioactivity to the water in the Receiving Basin for Offsite Fuel. The Draft EIS categorized the TRR fuel and EBR slugs as stable; however, based on public comment, further review by DOE, and recommendations by the Defense Nuclear Facilities Safety Board, DOE has categorized such fuel and slugs known to be failed as candidates for stabilization. The second group consists of americium and curium targets that the Draft EIS categorized as stable material but that the Final EIS categorizes as programmatic, based on several comments. Although these slugs are categorized as programmatic, they are stable and would not require a stabilization action. Chapter 4 contains analyses of reasonable alternatives for the stabilization of both of these material groups. DOE also revised the discussion of programmatic need to indicate that nuclear materials with a potential programmatic need would be candidates for stabilization even in the absence of a programmatic need; this change enabled DOE to remove the classified Appendix B that was part of the Draft EIS (and replace it with a discussion of radioactivity and health effects).

- Chapter 2 also contains additional discussion on how DOE selected the preferred alternatives for the nuclear materials. In the Final EIS, DOE has changed its selection of the preferred alternative for the Mark-16 and -22 fuels and the other aluminum-clad targets for the Final EIS to No Action; this preference allows additional time for DOE to conduct further comparisons of the relative costs of the various stabilization alternatives. Before making a decision on the stabilization method for the Mark-16 and -22 fuels and the other aluminum-clad targets, DOE will announce any revised preferred alternative and provide a 30-day waiting period before issuing a Record of Decision for these materials.

- DOE expanded the number of materials evaluated in various alternatives analyzed in the Draft EIS. Plutonium-242 now includes the Processing to Metal and the Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternatives. Similarly, the evaluations of neptunium and of americium and curium include Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative. The Mark-31 targets, the Mark-16 and -22 fuels, and the other aluminum-clad targets include an accelerated variation of the Improving Storage alternative based on a faster construction schedule.

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- DOE expanded the discussions in Chapters 2 and 4 and Appendixes D and E to provide additional information on the methodologies used to calculate risks and the basic assumptions used in these methodologies. DOE also included more detailed descriptions of the alternatives in Chapter 2 and reformatted Chapters 4 and 5 to better present the environmental impacts by alternative. Chapter 5 now contains the management scenarios that the Draft EIS presented in Chapter 4, which now contains only alternative-specific analysis. In addition, Chapter 5 describes a new management scenario, Minimum Processing. Finally, Chapter 2 contains more details on potential safety issues regarding facilities that DOE could use in stabilization actions.

- DOE revised Chapter 2 to include cost estimates for potential implementation schedules for the four management scenarios.

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- DOE added clarifying information in Chapter 1 to explain the rationale for using 10 years as the time period that DOE could take to make and implement decisions.

Purpose and Need for Action

The cessation in processing operations resulted in a large inventory of nuclear materials being caught in various stages of the historic production cycle (fabrication, irradiation, reprocessing, and recovery). These materials include irradiated and unirradiated reactor fuel, targets, and components; solutions containing dissolved nuclear materials and recovered isotopes in stainless-steel tanks; and product and scrap forms of metals or oxides in containers (cans, drums, etc.) typically used for temporary storage or shipment off the Site.

While DOE is evaluating various strategies for the long-term management and disposition of nuclear material (Section 1.6 describes these evaluations), the large inventory of nuclear materials at the SRS requires continued management. Some of the methods utilized to store these materials potentially pose risks to the environment or the safety and health of SRS workers or the public because, at the time DOE suspended the production cycle, many of these materials either were in a form or were TC stored in a manner that was intended for only a temporary period (e.g., 1 to 2 years). In some cases, the material's physical or chemical form poses a risk; in other cases, the material simply needs to be repackaged or moved to another location to ensure its safe storage. As the materials are currently stored, a number of accidents could result in the release of radioactivity to the environment and exposure of workers and the public. By taking action to alter the physical or chemical form of the materials or to alter the manner in which they are stored, DOE can reduce or eliminate the potential for their accidental release. Figure S-1 shows how the implementation of DOE's proposed actions could ultimately reduce public risk due to accidents; Appendix E contains the data to support this figure.

While reducing risk to the public, workers, and the environment is the primary purpose and need for the proposed action, some nuclear materials stored at the SRS contain special isotopes that could be required to support ongoing DOE programs. Although the need for strategic isotopes has been greatly diminished by the end of the Cold War, certain materials require additional processing or conversion to forms suitable both for continued safe storage and potential use.

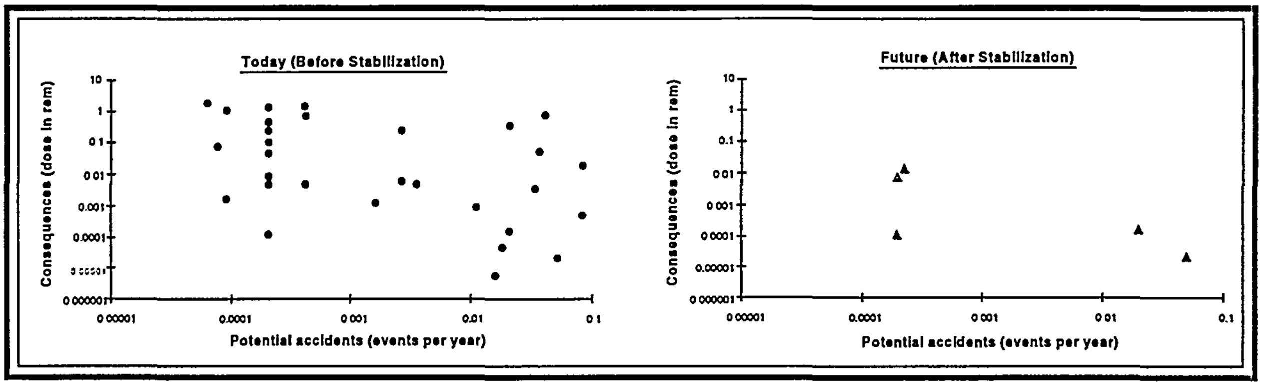

Figure S-1. Public risk profile before and after stabilization actions (assuming implementation of preferred alternatives).

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Categories of Nuclear Materials

Within the last 2 years, DOE completed two major studies to identify existing or potential environmental, safety, or health concerns associated with the storage of spent fuel or plutonium at DOE facilities nationwide (DOE 1994a,b). The studies identified a number of concerns associated with nuclear materials currently stored at the SRS. These materials include radioactive solutions stored in the chemical separations facilities, plutonium oxides and metals stored in vaults, and irradiated fuel and target assemblies stored in water-filled basins. In May 1994, the Defense Nuclear Facilities Safety Board recommended to the Secretary of Energy that DOE develop an integrated management plan to alleviate safety concerns associated with the materials at the SRS and other materials that remain from the nuclear weapons production cycle (DNFSB 1994). On the basis of the DOE evaluations and the Board's recommendation, DOE believes that it should consider actions necessary to ensure that these materials are placed in forms that are safe for interim storage. This EIS describes these materials as "candidates for stabilization."

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Materials that are candidates for stabilization are in forms (e.g., liquid) that present inherent management risks, are stored in facilities that were not designed for long-term storage (e.g., reactor disassembly basins), or both. In general, materials stored in liquid form are unsuitable for extended storage because of the greater potential for events (e.g., criticality) that could result in releases of radioactive materials to the environment and exposure to workers and the public. Certain solid materials represent similar concerns due to their chemical composition (reactive, corrosive, etc.), physical condition, or packaging composition. In most cases, these concerns stem from materials being kept in storage longer than the periods for which their packaging was designed. For example, fuel and targets stored in reactor disassembly basins have been there for as long as 6 years; in the past, such items were typically stored for approximately 6 months before processing. The extended

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TE storage of the fuel and targets in basins with poor water quality and limited filtration capability has produced surface corrosion that has affected the integrity of the cladding (i.e., the other layer of metal over the fissile material of a nuclear fuel element), resulting in continued releases of radioactivity to the surrounding water.

L14-3 DOE has evaluated the various activities that support its mission and has determined that there is a potential need for the plutonium-242, americium, curium, and neptunium-237 currently stored at the SRS in solutions, target assemblies, or slugs. Potential uses for these materials include support for such ongoing activities as the production of thermal power sources or special isotopes for medical applications and research. DOE has categorized these as "programmatic" materials.

TE DOE has evaluated the other nuclear materials at the SRS and believes that it can store them safely in their current forms and locations over the next 10 years. DOE has categorized these as "stable" materials. DOE does not propose any actions for these materials at this time except continued storage (i.e., No Action).

TE Table S-1 summarizes the nuclear materials at the SRS included in these categories. The "programmatic" and "candidates for stabilization" categories group the nuclear materials into subcategories due to differences in the physical or chemical composition of the materials and the corresponding interim management alternatives for each.

Alternatives

Table S-2 lists the alternatives that DOE considered in this EIS for each material category or subcategory. An open check mark indicates the preferred alternative for each material. The following paragraphs describe the alternatives:

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- **Continuing Storage (No Action).** DOE would continue to store the material in its current physical form. Continuing Storage is the preferred alternative for the stable materials and the americium and curium targets. DOE's preferred alternative at this time for the Mark-16 and -22 spent fuel and other aluminum-clad targets is Continuing Storage because DOE is in the process of conducting additional analyses of cost and implementation schedules for these materials. The additional analysis was prompted by public comments that DOE received on the Draft Interim Management of Nuclear Materials EIS about potential alternatives to technologies involving chemical dissolution. The analysis is likely to be complete within the next 2 to 4 months. DOE does not expect the additional analysis to affect the analyses of

Table S-1. SRS nuclear materials.[a]

Description	Quantity	Location(s)	
Stable			
Spent fuel	3,000 items	Receiving Basin for Offsite Fuel	
Unirradiated fuel, targets, reactor components, and scrap from fabrication operations	315,000 items	Buildings 305A, 313-M, 315-M, 320-M, 321-M, 322-M, and 341-M, K- and L-Reactor Assembly Areas	
Unirradiated fuel, targets, and reactor components	6,900 items	K- and L-Reactors	
Unirradiated and irradiated reactor components and control rods	420 items	C-, K-, L-, and P-Reactors	
Depleted uranium oxide	36,000 drums	R-Reactor, Buildings 221-1F, 221-12F, 221-21F, 221-22F, 707-R, 714-7N, 728-F, 730-F, and 772-7B	
Depleted uranium solutions	300,000 liters (78,000 gallons)	F-Canyon, F-Area Outside Facilities, and TNX	
Sources, standards, and samples	20,000 items	Sitewide	
Laboratory materials used in research and development	260 items	Savannah River Technology Center	
Programmatic			
Plutonium-242 solutions	13,000 liters (3,500 gallons)	H-Canyon	
Americium and curium solutions and targets	14,000 liters (3,800 gallons)	F-Canyon	
	65 assemblies	Receiving Basin for Offsite Fuel	L9-2
	60 slugs	P-Reactor disassembly basin	L14-4
	114 slugs	Receiving Basin for Offsite Fuel	
Neptunium solutions and targets	6,100 liters (1,600 gallons)	H-Canyon	
	9 targets	Building 321-M	
Candidates for Stabilization			
Plutonium-239 solutions	34,000 liters (9,000 gallons)	H-Canyon	
Highly enriched uranium solutions	228,000 liters (60,000 gallons)	H-Canyon and H-Area Outside Facilities	TE
Plutonium vault materials	2,800 packages	FB-Line, HB-Line, Building 772-F, Building 235-F, and SRTC	
Mark-31 targets	16,000 slugs	K-Reactor, L-Reactor, F-Canyon, and RBOF	
Mark-16 and Mark-22 fuels	1,900 assemblies	K-, L-, and P-Reactors and H-Canyon	
Other aluminum-clad targets	1,800 slugs and assemblies	K-, L-, and P-Reactors	
Failed TRR[b] fuel and EBR-II[c] slugs	82 canisters	Receiving Basin for Offsite Fuel	

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a. Appendix A contains a more comprehensive listing and description of these materials. Quantities of materials shown here are approximate. Quantities of radioactive solutions stored in tanks fluctuate due to natural evaporation and the addition of materials (e.g., nitric acid) to maintain chemistry within established parameters. Therefore, quantities listed in this table are approximate and might vary from quantities cited in previous DOE reports or studies. TC

b. Taiwan Research Reactor - 81 canisters.

c. Experimental Breeder Reactor-II - 1 canister.

Table S-2. Alternatives for the management of SRS nuclear materials.

	Alternatives							
Material	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[a]	Vitrification (F-Canyon)	Improving Storage	
Stable material	✔							TC
Plutonium-242	✓	✓	✔		✓	✓		L8-2
Americium and curium	✔[b]		✓		✓	✔[c]		L14-1
Neptunium	✓		✔		✓	✓		
H-Canyon plutonium-239 solutions	✓	✓	✔		✓	✓		
H-Canyon enriched uranium solutions	✓		✓	✔	✓			
Plutonium and uranium stored in vaults	✓	✔[d]	✔[d]		✓	✔[d]	✔[d]	TC
Mark-31 targets	✓	✔	✓		✓	✓	✓	
Mark-16 and -22 fuels	✔		✓	✓	✓		✓	
Other aluminum-clad targets	✔				✓		✓	
Failed TRR fuel and EBR-II slugs[e]	✓	✔	✓		✓	✓	✓	TC

✔ = Preferred alternative.

a. DWPF = Defense Waste Processing Facility.

b. Targets.

c. Solutions.

d. For the plutonium and uranium stored in vaults there are four preferred alternatives. DOE will base its choice of the appropriate alternative for a particular solid on results of the material inspection, as discussed in Section 2.3.3.

e. TRR = Taiwan Research Reactor; EBR-II = Experimental Breaker Reactor-II. TC potential environmental impacts for the management alternatives considered in this EIS. DOE believes that this additional analysis of cost and scheduling information for the Mark-16 and -22 spent fuel and other aluminum-clad targets furthers the intent and purposes of the National Environmental Policy Act by ensuring that the Department is as responsive as practicable to public comments and that the decisionmaker has all relevant information on which to base a decision. Before making a decision on the stabilization of the Mark-16 and -22 fuels and other aluminum-clad targets, DOE will announce any revised preferred alternative and provide a 30-day period before issuing a Record of Decision. TC

- **Processing to Metal.** DOE would use the existing F-Canyon and FB-Line facilities to dissolve materials containing significant amounts of plutonium-239 and convert the plutonium-239 to a metal. This would entail dissolving solids and purifying solutions before processing. The resulting plutonium metal would be packaged in a dry or inert atmosphere suitable for storage for as long as 50 years. The metal would be packaged and stored in either a modified facility (FB-Line or Building 235-F) or a new Actinide Packaging and Storage Facility in F-Area; this packaged metal would not be used in weapons. Processing to Metal is the preferred alternative for some of the plutonium-bearing materials stored in vaults, the Mark-31 targets, the failed Taiwan Research Reactor fuel, and the failed canister of Experimental Breeder Reactor-II slugs. TE

- **Processing to Oxide.** DOE would convert existing solutions containing neptunium-237 and plutonium-239 to oxides using either FB- or HB-Line, and would convert solutions containing highly enriched uranium to oxide using the Uranium Solidification Facility. Solid materials containing significant amounts of plutonium-239 or uranium-235 would be dissolved and the resulting solutions converted to an oxide in the same manner. Plutonium oxide would be packaged and stored in an existing vault facility (FB-Line, HB-Line, Building 235-F or 247-F), TE a modified facility (FB-Line or Building 235-F), or a new Actinide Packaging and Storage Facility in F-Area. Highly enriched uranium oxide would be stored in a vault in the Uranium Solidification Facility. Neptunium oxide would be packaged and stored in F-Canyon or an SRS vault. Processing to Oxide is the preferred alternative for plutonium-242 solutions, the H-Canyon plutonium-239 solutions, neptunium solutions, and some of the plutonium-bearing materials stored in vaults. TE

- **Blending Down to Low Enriched Uranium.** For those materials suitable for stabilization by this method, DOE would use depleted uranium to dilute highly enriched uranium to a low enrichment suitable for conversion to uranium oxide. Solid materials with enriched uranium (e.g., Mark-16 and -22 fuels) would be dissolved through traditional separation processing prior to this blending down activity; solutions of highly enriched uranium already being stored would be purified prior to the blending down. Low enriched uranium oxide would be stored in existing warehouses on the Site or in a new warehouse constructed in either F- or H-Area. TE The Blending Down to Low Enriched Uranium Alternative is preferred for the highly enriched uranium solutions.

- **Processing and Storage for Vitrification in the Defense Waste Processing Facility.** DOE would perform technical studies to determine the chemical adjustments required to enable the transfer of existing solutions containing significant amounts of fissile materials (e.g., plutonium-239, uranium-235) to the high-level waste tanks in F- or H-Area at the SRS. The solutions would subsequently be vitrified in the Defense Waste Processing Facility. Solid materials would be dissolved using existing chemical separations facilities (F- and H-Canyons) and the resulting solutions would be transferred and vitrified in the same manner. DOE TC included this alternative because of acknowledged stakeholder interest. However, Processing and Storage for Vitrification in the Defense Waste Processing Facility is not preferred for any of the material categories. Although technically feasible, the complexity of the alternative makes it unlikely for successful completion within the next 10 years.

- **Vitrification in F-Canyon.** DOE would modify part of the F-Canyon facility to install equipment to produce a glass composite, similar to that used for production in the Defense Waste Processing Facility. Existing solutions would be combined with molten borosilicate glass and poured into stainless-steel canisters. The canisters would be placed in storage in the canyon or in heavily shielded casks or vaults. Solid materials would be dissolved using existing F-Canyon or FB-Line facilities and the resulting solutions would be vitrified in a similar manner. The Vitrification in F-Canyon is the preferred alternative for americium and curium solutions, and those plutonium-bearing materials stored in vaults that contain corrosive or reactive compounds.

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- **Improving Storage.** DOE would repackage existing forms of solids. For small plutonium-bearing materials currently stored in vaults, DOE would modify the FB-Line facility or construct a new Actinide Packaging and Storage Facility to provide the capability to repackage such materials in a nonreactive atmosphere suitable for storage for as long as 50 years. For large irradiated materials (e.g., reactor fuel or targets), DOE would construct a new Dry Storage Facility with the capability to both repackage and store the materials. This would include the capability to can materials currently being stored in water in reactor disassembly basins. The Improving Storage alternative is a preferred alternative for those plutonium-bearing materials stored in vaults that do not contain corrosive or reactive compounds. TE

Affected Environment

The SRS occupies an area of approximately 800 square kilometers (300 square miles) adjacent to the Savannah River, primarily in Aiken and Barnwell Counties in South Carolina. The Site is approximately 40 kilometers (25 miles) southeast of Augusta, Georgia, and 32 kilometers (20 miles) south of Aiken, South Carolina. All alternatives (including No Action) would occur within existing industrial areas at the SRS except Improving Storage, which could involve the construction of a Dry Storage Facility for some materials (see Sections 2.3.4 through 2.3.7). TC

Comparison of Environmental Impacts Among the Alternatives

Tables S-3 through S-13 (which are at the end of this summary) list the potential environmental impacts associated with each of the nuclear materials for aspects of the environment that historically have held the most interest for the public. (Other aspects are considered in Chapter 4.) The tables list only the most significant chemical impacts for air and water resources. Radiological impacts for air and water resources are not listed specifically; however, those impacts are used to estimate latent cancer fatality impacts, which are listed. These tables also list estimated impacts (latent cancer fatalities) from the potential accident with the highest consequences associated with each alternative. Chapter 4 also contains tables of 10-year impacts, but these tables include additional parameters. For example, Tables S-3 through S-13 list the incremental contribution each alternative could make to the highest annual concentration of nitrogen oxide in the air, measured at the SRS boundary. The Chapter 4 tables list the highest 1-hour, 3-hour, 8-hour, 12-hour, 24-hour, weekly, or annual average concentration of the pollutants of concern in the air around the SRS. Tables S-3 through S-13 list only nitrogen oxide, because it is typically the nonradiological air pollutant of primary interest. In other words, the 10-year impact data shown in these tables are a subset of the more detailed 10-year impact data presented in Chapter 4. TC

To forecast the potential impacts of the implementation of each alternative over the next 10 years, DOE identified the various steps or "phases" required. DOE anticipates that the amount of environmental impacts would depend on the types of activities to be performed to manage the nuclear materials. For example, DOE expects different impacts from processing or repackaging operations than from maintaining nuclear material in a storage vault.

Appendix D contains annual estimates for the phases of each alternative. DOE used the data in Appendix D along with a schedule of potential activities to estimate the 10-year impacts for each alternative. Similarly, Appendix E analyzes environmental impacts from potential accidents associated with the phases of each alternative. The tables in Appendix E identify the accidents that pose either the greatest risk (considering both the likelihood and the consequences) or the greatest consequence (in latent cancer fatalities) in "bold" type. Again, DOE included the highest consequence accident for each alternative.

A decision to continue storing the materials considered in this EIS (i.e., No Action) probably would result in the smallest impacts for the 10-year period if the risk of accidents is disregarded. The reason for such a result is that the actions required to stabilize materials would entail some increased exposure and risk in comparison to No Action during the 10-year period. Over the long term, however, No Action could result in greater impacts than those that would occur from other alternatives, because of the increased possibility that continued changes in material chemistry or degradation of the functions and physical structure of the facilities containing the materials could result in releases to the environment and consequent worker exposures. Furthermore, DOE eventually would have to take some type of stabilization action, and the attendant risks and exposures from these actions would occur at that time.

TC

For alternatives that would be initiated but not completed within the 10-year period of analysis, DOE has provided information on the long-term impacts of the alternative under normal operating conditions. For example, alternatives involving vitrification in the Defense Waste Processing Facility would not be completed for many years due to the existing volume of high-level waste requiring vitrification. However, DOE has estimated the number of high-level waste canisters resulting from this alternative that would require placement in a geologic repository.

Only one alternative, Improving Storage, could involve the construction of a new facility outside an existing industrialized area (e.g., F-Area) of the SRS. For reactor fuel or targets, the Improving Storage Alternative would involve the construction of a new facility to dry the assemblies and package them for continued storage. For the purposes of determining environmental impacts, DOE assumed the new facility would be on a previously undisturbed site on the SRS. If practical, however, DOE would seek to minimize environmental impacts by siting the facility on a previously developed area of the SRS. DOE has considered two possible configurations for the Dry Storage Facility: a dry vault design and a dry cask design. There would be only slight differences in construction impacts from these two designs. For the use of either design, DOE anticipates little or no environmental impacts. In addition, DOE expects that the relative socioeconomic impact from Dry Storage Facility construction expenditures and construction employment on the region of influence around the SRS would be negligible because members of the current work force probably would perform the construction. TC

Several alternatives would require modifications to existing facilities. Most of these modifications would occur within existing buildings and facilities. For alternatives that would involve new facilities to package and store plutonium or uranium materials, DOE would construct the facilities within the already industrialized F- or H-Area. The new facility, which would be near existing nuclear facilities in those areas, would be a warehouse or concrete vault-type structure. Because construction would be confined to developed areas that have already been disturbed, DOE expects little or no environmental impacts to the following:

- Geological resources
- Ecological resources, including threatened or endangered species
- Cultural resources
- Aesthetics and scenic resources
- Noise
- Land use TE

Because any construction projects would be limited to modifying existing facilities or constructing warehouse or vault-type facilities (i.e., not complex major nuclear facilities), DOE anticipates that the existing SRS workforce would support these construction projects. Similarly, DOE would use the existing Site workforce to implement any of the alternatives considered. As a result, DOE expects negligible socioeconomic impacts from actions proposed in this EIS.

In addition to comparing alternatives to the environmental criteria listed in Tables S-3 through S-13, DOE considered the following factors related to the stabilization of nuclear materials:

- New facilities required
- Security and nonproliferation
- Implementation schedule
- Technology availability and technical feasibility
- Labor availability and core competency
- Aging facilities
- Minimum custodial care

TE These factors are representative of the issues addressed by the National Academy of Sciences in its study of the managed disposition of plutonium (NAS 1994), the Office of Technology Assessment plutonium study (OTA 1993), and comments received during the scoping period for this EIS.

TE In general, the alternatives that DOE identified as preferred (1) minimize the need for DOE to construct new facilities (those that are needed could be completed within 10 years), (2) rely on existing technology, (3) involve the use of existing personnel, and (4) minimize future custodial care for the materials. The preferred alternatives would also minimize continued reliance on aging facilities because DOE would consolidate nuclear materials into fewer and newer facilities.

TE Some additional weapons-usable material could result from actions proposed in this EIS. The amount would be a small fraction of the current SRS inventory and an even smaller fraction of that held at other DOE sites. All of the alternatives considered in this EIS would involve the use of facilities inside controlled industrial areas of the SRS, which are supported and protected by an armed guard force. DOE has committed to prohibit the use of plutonium-239 and weapons-usable highly enriched uranium separated or stabilized during the phaseout, shutdown, and cleanout of weapons complex facilities for nuclear explosive purposes (DOE 1994c).

Table S-3. Comparison of the potential environmental impacts of the alternatives for stable material.[a]

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Factors	Alternatives						
	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)	Vitrification (F-Canyon)	Improving Storage
Health effects of Normal Operations							
Radiological health effects (10-year totals):							
Offsite population latent cancer fatalities	0.0006	NA[b]	NA	NA	NA	NA	NA
Worker latent cancer fatalities	0.056	NA	NA	NA	NA	NA	NA
Health effects to offsite population from facility accidents[c] (projected latent cancer fatalities)	0.48	NA	NA	NA	NA	NA	NA
Risk (latent cancer fatalities per year)[d]	1.2×10^{-7}	NA	NA	NA	NA	NA	NA
Health effects from transportation (projected latent cancer fatalities)							
Incident-free (involved worker)	0.011	NA	NA	NA	NA	NA	NA
Accidents (offsite population)[e]	0.0000021	NA	NA	NA	NA	NA	NA
Air resources							
Nonradiological - Nitrogen oxide incremental concentration at SRS boundary (highest annual, micrograms per cubic meter)	0.0053	NA	NA	NA	NA	NA	NA
Water resources							
Lead (micrograms per liter) discharged to SRS streams	0	NA	NA	NA	NA	NA	NA
Utilities (10-year totals)							
Electricity usage (megawatt-hour)	400,000	NA	NA	NA	NA	NA	NA
Waste management (10-year totals)							
High-level liquid waste generation (million liters)	21	NA	NA	NA	NA	NA	NA
Equivalent DWPF canisters	40	NA	NA	NA	NA	NA	NA
Saltstone generation (cubic meters)	11,000	NA	NA	NA	NA	NA	NA
Transuranic waste generation (cubic meters)	20	NA	NA	NA	NA	NA	NA
Hazardous/mixed waste generation (cubic meters)	60	NA	NA	NA	NA	NA	NA
Low-level radioactive waste generation (cubic meters)	41,000	NA	NA	NA	NA	NA	NA

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TC

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a. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.

b. NA = Not applicable.

c. Assumes highly unlikely occurrence of maximum consequence accident.

d. Highest point estimate of risk during any phase of the alternative. The risk is calculated as the potential consequence of an accident (in latent cancer fatalities) multiplied by the probability per year of the accident occurring.

e. Maximum reasonably foreseeable latent cancer fatalities from medium probability accident based on the shipment of waste.

TE

Table S-4. Comparison of the potential environmental impacts of the alternatives for plutonium-242.[a]

TE

Factors	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[b]	Vitrification (F-Canyon)	Improving Storage
	Alternatives						
Health effects of Normal Operations							
Radiological health effects (10-year totals):							
Offsite population latent cancer fatalities	0.00025	0.00024	0.0017	NA[c]	0.00016	0.0017	NA
Worker latent cancer fatalities	0.0052	0.035	0.024	NA	0.0035	0.027	NA
Health effects to offsite population from facility accidents[d] (projected latent cancer fatalities)	6.5	6.5	6.5	NA	38	6.5	NA
Risk (latent cancer fatalities per year)[e]	0.000015	0.000015	0.000015	NA	0.000015	0.000015	NA
Health effects from transportation[f] (projected latent cancer fatalities)							
Incident-free (involved worker)	0.0017[g]	0.0019	0.0011	NA	0.0011[g]	0.0012	NA
Accidents (offsite population)[h]	0.0000021	2	2	NA	0.0000021	2	NA
Air resources							
Nonradiological - Nitrogen oxide incremental concentration at SRS boundary (highest annual, micrograms per cubic meter)	0.012	0.140	0.033	NA	0.033	0.11	NA
Water resources							
Lead (micrograms per liter) discharged to SRS streams	3.2	2.7	2.7	NA	3.2	2.8	NA
Utilities (10-year totals)							
Electricity usage (megawatt-hour)	133,000	127,000	41,000	NA	85,000	42,000	NA
Waste management (10-year totals)							
High-level liquid waste generation (million liters)	1.2	1.2	0.12	NA	0.77	0.16	NA
Equivalent DWPF canisters	20	21	2	NA	23	3	NA
Saltstone generation (cubic meters)	3,300	3,200	330	NA	2,100	420	NA
Transuranic waste generation (cubic meters)	0	21	56	NA	0	61	NA
Hazardous/mixed waste generation (cubic meters)	0	0	60	NA	0	60	NA
Low-level radioactive waste generation (cubic meters)	5,600	6,500	4,300	NA	3,500	4,700	NA

TC

TC

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a. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.
b. The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.
c. NA = Not applicable.
d. Assumes highly unlikely occurrence of maximum consequence accident.
e. Highest point estimate of risk during any phase of the alternative. The risk is calculated as the potential consequence of an accident (in latent cancer fatalities) multiplied by the probability per year of the accident occurring.
f. Includes transportation of associated radioactive waste.
g. Waste transportation only.
h. Maximum reasonably foreseeable latent cancer fatalities from medium probability accident based on the shipment of waste.

TE

Table S-5. Comparison of the potential environmental impacts of the alternatives for americium and curium.[a,b]

TE

	Alternatives						
Factors	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[c]	Vitrification (F-Canyon)	Improving Storage
Health effects of Normal Operations							
Radiological health effects (10-year totals):							
Offsite population latent cancer fatalities	0.00035	NA[d]	0.0012	NA	0.00041	0.00050	NA
Worker latent cancer fatalities	0.034	NA	0.128	NA	0.044	0.052	NA
Health effects to offsite population from facility accidents[c] (projected latent cancer fatalities)	3.1	NA	6.5	NA	38	6.5	NA
Risk (latent cancer fatalities per year)[f]	5.7×10^{-8}	NA	0.000015	NA	0.000015	0.000015	NA
Health effects from transportation[g] (projected latent cancer fatalities)							
Incident-free (involved worker)	0.0022	NA	0.0041	NA	0.0018[h]	0.0002	NA
Accidents (offsite population)[i]	0.0000021	NA	0.0000021	NA	0.0000021	0.0000021	NA
Air resources							
Nonradiological - Nitrogen oxide incremental concentration at SRS boundary (highest annual, micrograms per cubic meter)	0.033	NA	0.28	NA	0.23	0.28	NA
Water resources							
Lead (micrograms per liter) discharged to SRS streams	6.1	NA	6.0	NA	6.1	5.8	NA
Utilities (10-year totals)							
Electricity usage (megawatt-hour)	140,000	NA	181,000	NA	110,000	100,000	NA
Waste management (10-year totals)							
High-level liquid waste generation (million liters)	1.7	NA	6.9	NA	2.6	2.7	NA
Equivalent DWPF canisters	30	NA	140	NA	50	54	NA
Saltstone generation (cubic meters)	3,600	NA	18,000	NA	6,100	6,900	NA
Transuranic waste generation (cubic meters)	0	NA	0	NA	0	0	NA
Hazardous/mixed waste generation (cubic meters)	0	NA	0	NA	0	0	NA
Low-level radioactive waste generation (cubic meters)	7,700	NA	9,200	NA	6,300	5,100	NA

TC

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a. Impact data for each alternative include management of both solution and targets.
b. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.
c. The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.
d. NA = Not applicable.
e. Assumes highly unlikely occurrence of maximum consequence accident.
f. Highest point estimate of risk during any phase of the alternative. The risk is calculated as the potential consequence of an accident (in latent cancer fatalities) multiplied by the probability per year of the accident occurring.
g. Includes transportation of associated radioactive waste.
h. Waste transportation only.
i. Maximum reasonably foreseeable latent cancer fatalities from medium probability accident based on the shipment of waste.

TE

Table S-6. Comparison of the potential environmental impacts of the alternatives for neptunium.[a,b]

TE

Factors	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[c]	Vitrification (F-Canyon)	Improving Storage
Health effects of Normal Operations							
Radiological health effects (10-year totals):							
Offsite population latent cancer fatalities	0.00027	NA[d]	0.028	NA	0.0047	0.00023	NA
Worker latent cancer fatalities	0.006	NA	0.052	NA	0.0056	0.020	NA
Health effects to offsite population from facility accidents[e] (projected latent cancer fatalities)	4.1	NA	4.1	NA	38	6.5	NA
Risk (latent cancer fatalities per year)[f]	0.0000036	NA	0.0000036	NA	0.0000036	0.000015	NA
Health effects from transportation[g] (projected latent cancer fatalities)							
Incident-free (involved worker)	0.0017[h]	NA	0.0028	NA	0.0014[h]	0.0013	NA
Accidents (offsite population)[i]	0.0000021	NA	2.0	NA	0.0000021	0.0000021	NA
Air resources							
Nonradiological - Nitrogen oxide incremental concentration at SRS boundary (highest annual, micrograms per cubic meter)	0.019	NA	0.10	NA	0.083	0.11	NA
Water resources							
Lead (micrograms per liter) discharged to SRS streams	3.0	NA	3.0	NA	3.0	3.0	NA
Utilities (10-year totals)							
Electricity usage (megawatt-hour)	142,000	NA	149,000	NA	93,000	126,000	NA
Waste management (10-year totals)							
High-level liquid waste (million liters)	1.3	NA	4.2	NA	1.5	1.0	NA
Equivalent DWPF canisters	20	NA	37	NA	27	16	NA
Saltstone generation (cubic meters)	3,600	NA	11,000	NA	4,000	2,800	NA
Transuranic waste generation (cubic meters)	0	NA	160	NA	0	0	NA
Hazardous/mixed waste generation (cubic meters)	0	NA	200	NA	0	0	NA
Low-level radioactive waste generation (cubic meters)	5,700	NA	6,400	NA	3,800	4,600	NA

TC

TC

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a. Impact data for each alternative include management of both solution and targets.
b. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.
c. The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.
d. NA = Not applicable.
e. Assumes highly unlikely occurrence of maximum consequence accident.
f. Highest point estimate of risk during any phase of the alternative. The risk is calculated as the potential consequence of an accident (in latent cancer fatalities) multiplied by the probability per year of the accident occurring.
g. Includes transportation of associated radioactive waste.
h. Waste transportation only.
i. Maximum reasonably foreseeable latent cancer fatalities from medium probability accident based on the shipment of waste.

TE

Table S-7. Comparison of the potential environmental impacts of the alternatives for H-Canyon plutonium-239 solutions.[a]

TE

Factors	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[b]	Vitrification (F-Canyon)	Improving Storage
	Alternatives						
Health effects of Normal Operations							
Radiological health effects (10-year totals):							
Offsite population latent cancer fatalities	0.00025	0.00025	0.0018	NA[c]	0.041	0.00023	NA
Worker latent cancer fatalities	0.0052	0.044	0.026	NA	0.02	0.021	NA
Health effects to offsite population from facility accidents[d] (projected latent cancer fatalities)	4.1	6.5	4.1	NA	38	6.5	NA
Risk (latent cancer fatalities per year)[e]	0.0000036	0.000015	0.0000036	NA	0.0000036	0.000015	NA
Health effects from transportation[f] (projected latent cancer fatalities)							
Incident-free (involved worker)	0.0017[g]	0.0022[g]	0.0017	NA	0.0036[g]	0.0014[g]	NA
Accidents (offsite population)[h]	0.0000021	2.0	2.0	NA	0.0000021	0.0000021	NA
Air resources							
Nonradiological - Nitrogen oxide incremental concentration at SRS boundary (highest annual, micrograms per cubic meter)	0.012	0.14	0.033	NA	0.083	0.11	NA
Water resources							
Lead (micrograms per liter) discharged to SRS streams	3.2	3.3	2.9	NA	3.0	3.2	NA
Utilities (10-year totals)							
Electricity usage (megawatt-hour)	133,000	135,000	89,000	NA	151,000	124,000	NA
Waste management (10-year totals)							
High-level liquid waste (million liters)	1.2	1.3	0.55	NA	6.8	1.0	NA
Equivalent DWPF canisters	20	24	9	NA	190	17	NA
Saltstone generation (cubic meters)	3,300	3,500	1,500	NA	19,000	2,700	NA
Transuranic waste generation (cubic meters)	0	32	56	NA	0	0	NA
Hazardous/mixed waste generation (cubic meters)	0	0	63	NA	0	0	NA
Low-level radioactive waste generation (cubic meters)	5,600	7,500	6,300	NA	6,400	4,800	NA

TC

TC

[a] To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.

[b] The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.

[c] NA = Not applicable.

[d] Assumes highly unlikely occurrence of maximum consequence accident.

[e] Highest point estimate of risk during any phase of the alternative. The risk is calculated as the potential consequence of an accident (in latent cancer fatalities) multiplied by the probability per year of the accident occurring.

[f] Includes transportation of associated radioactive waste.

[g] Waste transportation only.

[h] Maximum reasonably foreseeable latent cancer fatalities from medium probability accident based on the shipment of waste.

TE

Table S-8. Comparison of the potential environmental impacts of the alternatives for H-Canyon enriched uranium solutions.[a]

TE

Factors	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[b]	Vitrification (F-Canyon)	Improving Storage
Health effects of Normal Operations							
Radiological health effects (10-year totals):							
Offsite population latent cancer fatalities	0.00038	NA[c]	0.0034	0.009	0.0003	NA	NA
Worker latent cancer fatalities	0.0092	NA	0.028	0.0072	0.0072	NA	NA
Health effects to offsite population from facility accidents[d] **(projected latent cancer fatalities)**	0.14	NA	0.14	0.14	38	NA	NA
Risk (latent cancer fatalities per year)[e]	9.6×10^{-7}	NA	9.6×10^{-7}	9.6×10^{-7}	0.0000036	NA	NA
Health effects from transportation[f] **(projected latent cancer fatalities)**							
Incident-free (involved worker)	0.002[g]	NA	0.0005[g]	0.00089	0.0016[g]	NA	NA
Accidents (offsite population)[h]	0.0000021	NA	0.0000021	0.0000021	0.0000021	NA	NA
Air resources							
Nonradiological - Nitrogen oxide incremental concentration at SRS boundary (highest annual, micrograms per cubic meter)	0.053	NA	0.083	0.083	0.053	NA	NA
Water resources							
Lead (micrograms per liter) discharged to SRS streams	3.0	NA	3.0	3.0	3.0	NA	NA
Utilities (10-year totals)							
Electricity usage (megawatt-hour)	180,000	NA	40,000	42,000	140,000	NA	NA
Waste management (10-year totals)							
High-level liquid waste (million liters)	1.8	NA	0.72	1.7	1.4	NA	NA
Equivalent DWPF canisters	30	NA	7	17	130	NA	NA
Saltstone generation (cubic meters)	5,000	NA	2,000	4,800	3,900	NA	NA
Transuranic waste generation (cubic meters)	0	NA	0	0	0	NA	NA
Hazardous/mixed waste generation (cubic meters)	0	NA	0	0	0	NA	NA
Low-level radioactive waste generation (cubic meters)	6,300	NA	1,200	1,600	4,800	NA	NA

TC

TC

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a. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.
b. The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.
c. NA = Not applicable.
d. Assumes highly unlikely occurrence of maximum consequence accident.
e. Highest point estimate of risk during any phase of the alternative. The risk is calculated as the potential consequence of an accident (in latent cancer fatalities) multiplied by the probability per year of the accident occurring.
f. Includes transportation of associated radioactive waste.
g. Waste transportation only.
h. Maximum reasonably foreseeable latent cancer fatalities from medium probability accident based on the shipment of waste.

TE

Table S-9. Comparison of the potential environmental impacts of the alternatives for plutonium and uranium stored in vaults.[a]

TE

Factors	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[b]	Vitrification (F-Canyon)	Improving Storage
Health effects of Normal Operations							
Radiological health effects (10-year totals):							
Offsite population latent cancer fatalities	0.00011	0.07	0.07	NA[c]	0.07	0.07	0.00024
Worker latent cancer fatalities	0.056	0.18	0.18	NA	0.11	0.18	0.16
Health effects to offsite population from facility accidents[d] **(projected latent cancer fatalities)**	0.31	4.1	4.5	NA	38	4.1	0.62
Risk (latent cancer fatalities per year)[e]	1×10^{-8}	0.0000036	0.0000022	NA	8.9×10^{-7}	0.0000036	2×10^{-8}
Health effects from transportation[f] **(projected latent cancer fatalities)**							
Incident-free (involved worker)	0.0052	0.0091	0.0091	NA	0.0077	0.0091	0.0062
Accidents (offsite population)[g]	2.0	2.0	2.0	NA	2.0	2.0	2.0
Air resources							
Nonradiological - Nitrogen oxide incremental concentration at SRS boundary (highest annual, micrograms per cubic meter)	0.0095	0.14	0.14	NA	0.13	0.14	0.031
Water resources							
Lead (micrograms per liter) discharged to SRS streams	3.1	3.1	3.1	NA	3.1	3.1	4.3
Utilities (10-year totals)							
Electricity usage (megawatt-hour)	147,000	190,000	190,000	NA	210,000	190,000	77,000
Waste management (10-year totals)							
High-level liquid waste (million liters)	0	8.2	8.2	NA	8.2	8.2	0
Equivalent DWPF canisters	0	61	61	NA	2,400	61	0
Saltstone generation (cubic meters)	0	22,000	22,000	NA	22,000	22,000	0
Transuranic waste generation (cubic meters)	810	1,300	1,300	NA	900	1,300	1,000
Hazardous/mixed waste generation (cubic meters)	970	1,400	1,400	NA	1,100	1,400	960
Low-level radioactive waste generation (cubic meters)	19,000	24,000	24,000	NA	19,000	24,000	23,000

TC

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a. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts. In addition, DOE calculated impacts assuming that all plutonium and uranium stored in vaults are using the listed alternative to bound the impact estimate.
b. The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.
c. NA = Not applicable.
d. Assumes highly unlikely occurrence of maximum consequence accident.
e. Highest point estimate of risk during any phase of the alternative. The risk is calculated as the potential consequence of an accident (in latent cancer fatalities) multiplied by the probability per year of the accident occurring.
f. Includes transportation of associated radioactive waste.
g. Maximum reasonably foreseeable latent cancer fatalities from medium probability accident based on the shipment of waste.

TE

Table S-10. Comparison of the potential environmental impacts of the alternatives for Mark-31 targets.[a]

TE

Factors	Alternatives							
							Improving Storage	
	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[b]	Vitrification (F-Canyon)	Traditional construction schedule	Accelerated construction schedule
Health effects of Normal Operations								
Radiological health effects (10-year totals):								
Offsite population latent cancer fatalities	0.00006	0.00025	0.00023	NA[c]	0.00043	0.00032	0.00006	0.000032
Worker latent cancer fatalities	0.0056	0.084	0.072	NA	0.044	0.1	0.0056	0.0076
Health effects to offsite population from facility accidents[d] (projected latent cancer fatalities)	0.0089	6.5	6.5	NA	38	6.5	0.0089	0.0089
Risk (latent cancer fatalities per year)[e]	4.9×10^{-9}	0.000015	0.000015	NA	0.000015	0.000015	4.9×10^{-9}	4.9×10^{-9}
Health effects from transportation[f] (projected latent cancer fatalities)								
Incident-free (involved worker)	0.0073[g]	0.0049	0.0048	NA	0.0063	0.0053	0.0073	0.0053
Accidents (offsite population)[h]	0.0000021	2.0	2.0	NA	0.0000021	2.0	0.0000021	0.0000021
Air resources								
Nonradiological - Nitrogen oxide incremental concentration at SRS boundary (highest annual, micrograms per cubic meter)	0	0.28	0.28	NA	0.23	0.34	0	0
Water resources								
Lead (micrograms per liter) discharged to SRS streams	0	3.4	3.5	NA	6.1	3.9	0	0
Utilities (10-year totals)								
Electricity usage (megawatt-hour)	14	64,000	51,000	NA	44,000	71,000	14	1,400
Waste management (10-year totals)								
High-level liquid waste generation (million liters)	1.2	2.1	1.9	NA	3.7	2.6	1.2	0.87
Equivalent DWPF canisters	28	43	41	NA	170	53	28	18
Saltstone generation (cubic meters)	3,200	5,700	5,200	NA	10,000	7,000	3,200	2,300
Transuranic waste generation (cubic meters)	0	77	62	NA	0	93	0	0
Hazardous/mixed waste generation (cubic meters)	50	16	20	NA	34	16	50	35
Low-level radioactive waste generation (cubic meters)	29,000	18,000	18,000	NA	22,000	19,000	29,000	21,000

TC

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a. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.

b. The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.

c. NA = Not applicable.

d. Assumes highly unlikely occurrence of maximum consequence accident.

e. Highest point estimate of risk during any phase of the alternative. The risk is calculated as the potential consequence of an accident (in latent cancer fatalities) multiplied by the probability per year of the accident occurring.

f. Includes transportation of associated radioactive waste.

g. Waste transportation only.

h. Maximum reasonably foreseeable latent cancer fatalities from medium probability accident based on the shipment of waste.

TE

Table S-11. Comparison of the potential environmental impacts of the alternatives for Mark-16 and -22 fuels.[a]

TE

	Alternatives							
							Improving Storage	
Factors	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[b]	Vitrification (F-Canyon)	Traditional construction schedule	Accelerated construction schedule
Health effects of Normal Operations								
Radiological health effects (10-year totals):								
Offsite population latent cancer fatalities	0.000016	NA[c]	0.034	0.041	0.0008	NA	0.000016	0.000011
Worker latent cancer fatalities	0.0028	NA	0.08	0.026	0.088	NA	0.0028	0.0068
Health effects to offsite population from facility accidents[d] (projected latent cancer fatalities)	0.0089	NA	4.1	4.1	38	NA	0.0089	0.0089
Risk (latent cancer fatalities per year)[e]	4.9×10^{-9}	NA	0.0000036	0.0000036	0.0000036	NA	4.9×10^{-9}	4.9×10^{-9}
Health effects from transportation[f] (projected latent cancer fatalities)								
Incident-free (involved worker)	0.0037[g]	NA	0.0054	0.0063	0.0097	NA	0.0038	0.0019
Accidents (offsite population)[h]	0.0000021	NA	0.0000021	0.0000021	0.0000021	NA	0.0000021	0.0000021
Air resources								
Nonradiological - Nitrogen oxide incremental concentration at SRS boundary (highest annual, micrograms per cubic meter)	0	NA	0.083	0.083	0.23	NA	0	0
Water resources								
Lead (micrograms per liter) discharged to SRS streams	0	NA	3	3	6.1	NA	0	0
Utilities (10-year totals)								
Electricity usage (megawatt-hour)	10	NA	79,000	83,000	89,000	NA	10	2,800
Waste management (10-year totals)								
High-level liquid waste generation (million liters)	0.57	NA	5.6	7.3	6.8	NA	0.57	0.37
Equivalent DWPF canisters	10	NA	49	68	1,000	NA	10	5
Saltstone generation (cubic meters)	1,600	NA	15,000	20,000	19,000	NA	1,600	800
Transuranic waste generation (cubic meters)	0	NA	0	0	0	NA	0	0
Hazardous/mixed waste generation (cubic meters)	20	NA	22	28	44	NA	20	10
Low-level radioactive waste generation (cubic meters)	15,000	NA	16,000	20,000	32,000	NA	15,000	7,700

TC

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a. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.
b. The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.
c. NA = Not applicable.
d. Assumes highly unlikely occurrence of maximum consequence accident.
e. Highest point estimate of risk during any phase of the alternative. The risk is calculated as the potential consequence of an accident (in latent cancer fatalities) multiplied by the probability per year of the accident occurring.
f. Includes transportation of associated radioactive waste.
g. Waste transportation only.
h. Maximum reasonably foreseeable latent cancer fatalities from medium probability accident based on the shipment of waste.

TE

Table S-12. Comparison of the potential environmental impacts of the alternatives for other aluminum-clad targets.[a]

TE

Factors	Alternatives							
							Improving Storage	
	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[b]	Vitrification (F-Canyon)	Traditional construction schedule	Accelerated construction schedule
Health effects of Normal Operations								
Radiological health effects (10-year totals):								
Offsite population latent cancer fatalities	0.0000049	NA[c]	NA	NA	0.0034	NA	0.0000049	0.0000031
Worker latent cancer fatalities	0.00084	NA	NA	NA	0.0018	NA	0.00084	0.0018
Health effects to offsite population from facility accidents[d] (projected latent cancer fatalities)	0.0089	NA	NA	NA	38	NA	0.0089	0.0089
Risk (latent cancer fatalities per year)[e]	4.9×10^{-9}	NA	NA	NA	0.0000036	NA	4.9×10^{-9}	4.9×10^{-9}
Health effects from transportation[f] (projected latent cancer fatalities)								
Incident-free (involved worker)	0.00105[g]	NA	NA	NA	0.00072	NA	0.0011	0.00057
Accidents (offsite population)[h]	0.0000021	NA	NA	NA	0.0000021	NA	0.0000021	0.0000021
Air resources								
Nonradiological - Nitrogen oxide incremental concentration at SRS boundary (highest annual, micrograms per cubic meter)	0	NA	NA	NA	0.083	NA	0	0
Water resources								
Lead (micrograms per liter) discharged to SRS streams	0	NA	NA	NA	3	NA	0	0
Utilities (10-year totals)								
Electricity usage (megawatt-hour)	10	NA	NA	NA	5,900	NA	10	720
Waste management (10-year totals)								
High-level liquid waste generation (million liters)	0.14	NA	NA	NA	0.59	NA	0.14	0.09
Equivalent DWPF canisters	0	NA	NA	NA	15	NA	0	0
Saltstone generation (cubic meters)	390	NA	NA	NA	1,600	NA	390	200
Transuranic waste generation (cubic meters)	0	NA	NA	NA	0	NA	0	0
Hazardous/mixed waste generation (cubic meters)	10	NA	NA	NA	4	NA	10	5
Low-level radioactive waste generation (cubic meters)	4,200	NA	NA	NA	2,300	NA	4,200	2,300

TC

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a. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.

b. The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.

c. NA = Not applicable.

d. Assumes highly unlikely occurrence of maximum consequence accident.

e. Highest point estimate of risk during any phase of the alternative. The risk is calculated as the potential consequence of an accident (in latent cancer fatalities) multiplied by the probability per year of the accident occurring.

f. Includes transportation of associated radioactive waste.

g. Waste transportation only.

h. Maximum reasonably foreseeable latent cancer fatalities from medium probability accident based on the shipment of waste.

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Table S-13. Comparison of the potential environmental impacts of the alternatives for Taiwan Research Reactor fuel and Experimental Breeder Reactor-II slugs.[a,b]

	Alternatives							
							Improving Storage	
Factors	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[c]	Vitrification (F-Canyon)	Traditional construction schedule	Accelerated construction schedule
Health effects of Normal Operations								
Radiological health effects (10-year totals):								
Offsite population latent cancer fatalities	0.000005	0.0002	0.0002	NA[d]	0.00017	0.00027	0.000005	0.000005
Worker latent cancer fatalities	0.0056	0.072	0.072	NA	0.021	0.088	0.0056	0.0084
Health effects to offsite population from facility accidents[e] (projected latent cancer fatalities)	(f)	6.5	6.5	NA	38	6.5	0.0089	0.0089
Risk (latent cancer fatalities per year)[g]	6.5×10^{-9}	0.000015	0.000015	NA	0.000015	0.000015	6.5×10^{-9}	6.5×10^{-9}
Health effects from transportation[h] (projected latent cancer fatalities)								
Incident-free (involved worker)	0.00033[i]	0.0034	0.0028	NA	0.00080	0.0037	0.00032	0.00032
Accidents (offsite population)[j]	0.0000021	2.0	2.0	NA	0.0000021	2.0	0.0000021	0.0000021
Air resources								
Nonradiological - Nitrogen oxide incremental concentration at SRS boundary (highest annual, micrograms per cubic meter)	0	0.28	0.28	NA	0.23	0.34	0	0
Water resources								
Lead (micrograms per liter) discharged to SRS streams	2.7	3.0	3.1	NA	3.3	3.4	2.7	2.8
Utilities (10-year totals)								
Electricity usage (megawatt-hour)	11,000	62,000	55,000	NA	27,000	69,000	11,000	8,600
Waste management (10-year totals)								
High-level liquid waste generation (million liters)	0.4	1.6	1.7	NA	1.5	2.1	0.4	0.3
Equivalent DWPF canisters	0	32	32	NA	120	41	0	0
Saltstone generation (cubic meters)	110	4,200	4,300	NA	3,300	5,500	110	77
Transuranic waste generation (cubic meters)	0	67	62	NA	0	82	0	0
Hazardous/mixed waste generation (cubic meters)	0	7	4	NA	0	6	0	0
Low-level radioactive waste generation (cubic meters)	1,300	12,000	9,600	NA	1,800	13,000	1,300	1,200

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a. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.

b. DOE derived the impact data for the TRR targets and the EBR-II slugs assuming it would have to stabilize all the material; this bounds the impacts in the event of additional TRR and EBR-II material failures.

c. The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.

d. NA = Not applicable.

e. Assumes highly unlikely occurrence of maximum consequence accident.

f. Data not available.

g. Highest point estimate of risk during any phase of the alternative. The risk is calculated as the potential consequence of an accident (in latent cancer fatalities) multiplied by the probability per year of the accident occurring.

h. Includes transportation of associated radioactive waste.

i. Waste transportation only.

j. Maximum reasonably foreseeable latent cancer fatalities from medium probability accident based on the shipment of waste.

CHAPTER 1. PURPOSE AND NEED FOR ACTION

1.1 Introduction and Background

The U.S. Atomic Energy Commission (AEC), a predecessor agency of the Department of Energy (DOE), established the Savannah River Site (SRS) in the early 1950s. The SRS occupies an area of approximately 800 square kilometers (300 square miles) adjacent to the Savannah River, primarily in Aiken and Barnwell Counties in South Carolina. The Site is approximately 40 kilometers (25 miles) southeast of Augusta, Georgia, and 32 kilometers (20 miles) south of Aiken, South Carolina (Figure 1-1). Figure 1-2 shows the locations of the principal SRS facilities. | TE

The SRS mission for the past 40 years has been the production of special radioactive isotopes to support national programs. In this regard, the primary SRS mission was the production of strategic | TE isotopes (plutonium-239 and tritium) used in the development and production of nuclear weapons for national defense. The Site produced other special isotopes (e.g., californium-252, plutonium-238, americium-241) to support research in nuclear medicine, space exploration, and commercial applications. To produce the isotopes, DOE fabricated selected materials into metal targets and irradiated them in the SRS nuclear reactors. The targets and reactor fuel were dissolved in acid and | TE the special isotopes were chemically separated and converted to a solid form, either an oxide powder or a metal. The oxide or metal was fabricated into a usable form at the SRS or at other DOE sites. The final form of the material depended on the application (nuclear weapon component, encapsulated medical source, power source, etc.). Figure 1-3 shows the historic SRS production cycle.

Due to the large-scale chemical separation capabilities at the SRS, materials containing significant quantities of plutonium-239, uranium-235, and other special isotopes were shipped to the Site for | TE processing and recovery. The materials were in a wide variety of physical shapes and forms, including (1) small encapsulated plutonium sources returned from use by national laboratories and domestic universities; (2) cans or drums of scrap metals and oxides from weapons manufacturing operations at other DOE sites; (3) irradiated metal fuel rods, tubes, plates, or assemblies from experimental DOE reactors, university research reactors, and foreign research reactors; and (4) cans, bottles, or drums containing residues or samples used in laboratory experiments at other DOE sites. All the materials were stored until they could be dissolved and processed in the chemical separations facilities. The small sources, scrap metals, oxides, residues, and samples were typically stored in cans, bottles, or drums in safeguarded concrete vaults. The irradiated fuel and targets were stored under water in metal racks or buckets. The offsite materials were typically processed in conjunction with | TE

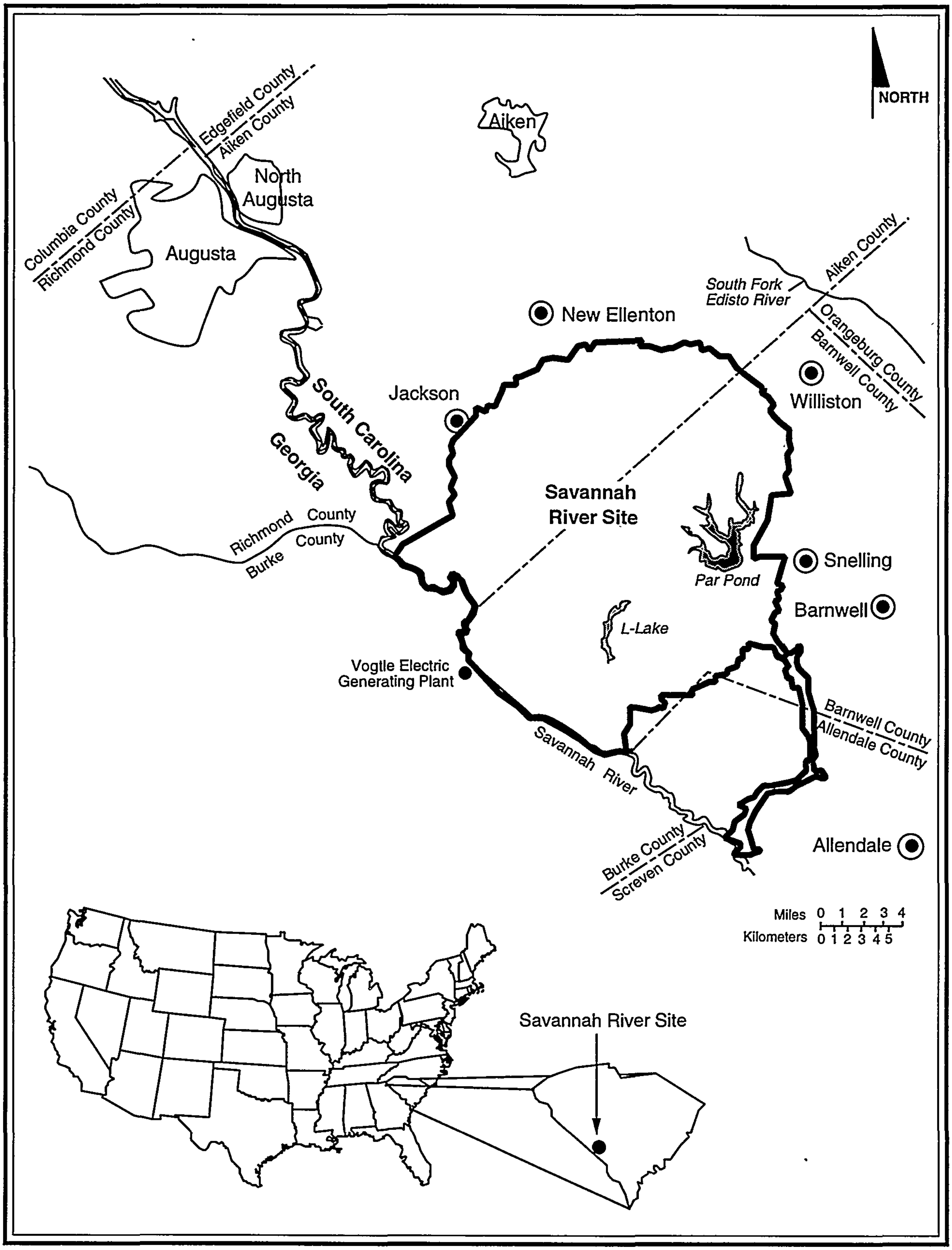

Figure 1-1. Location of the Savannah River Site.

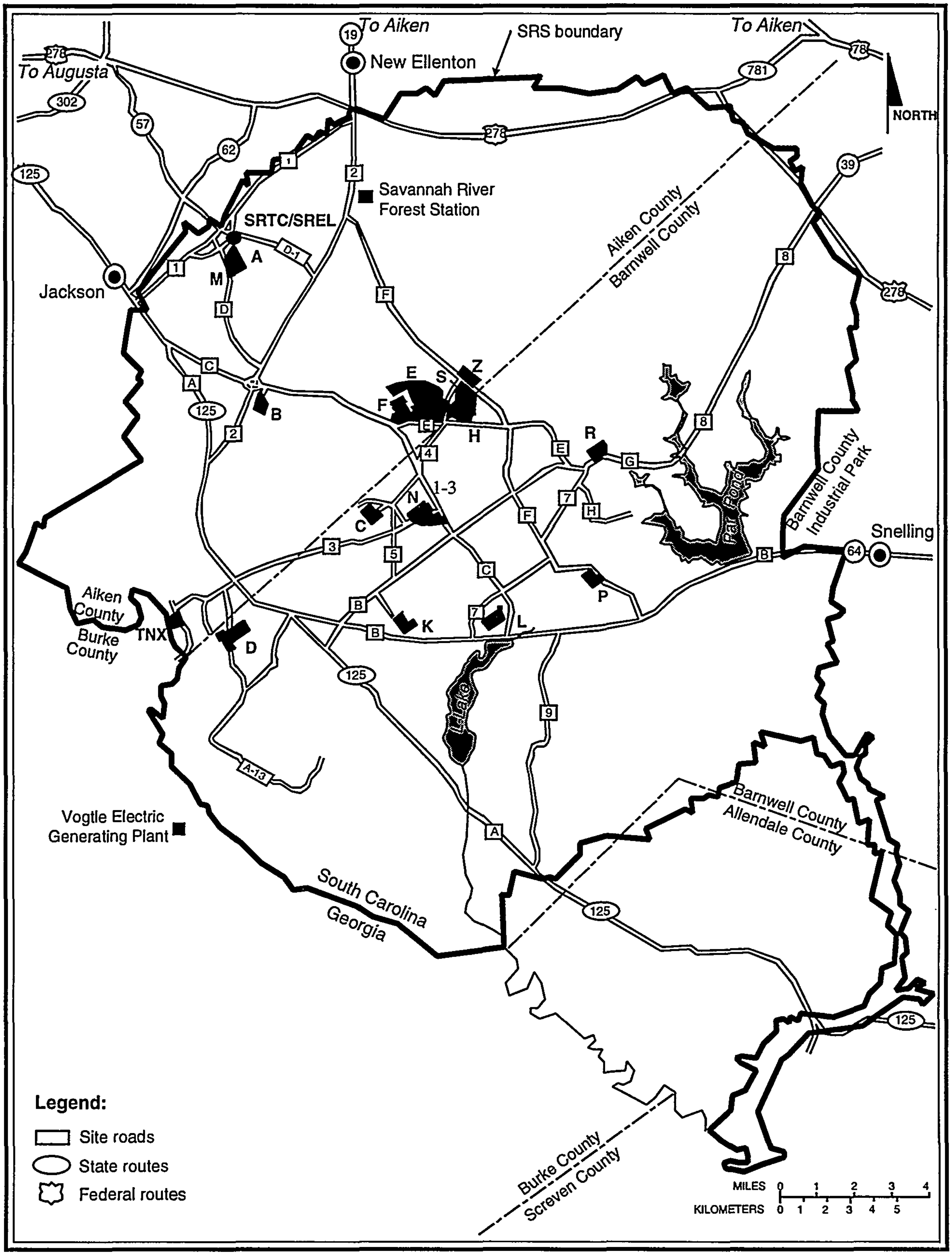

Figure 1-2. Locations of principal facilities at the Savannah River Site.

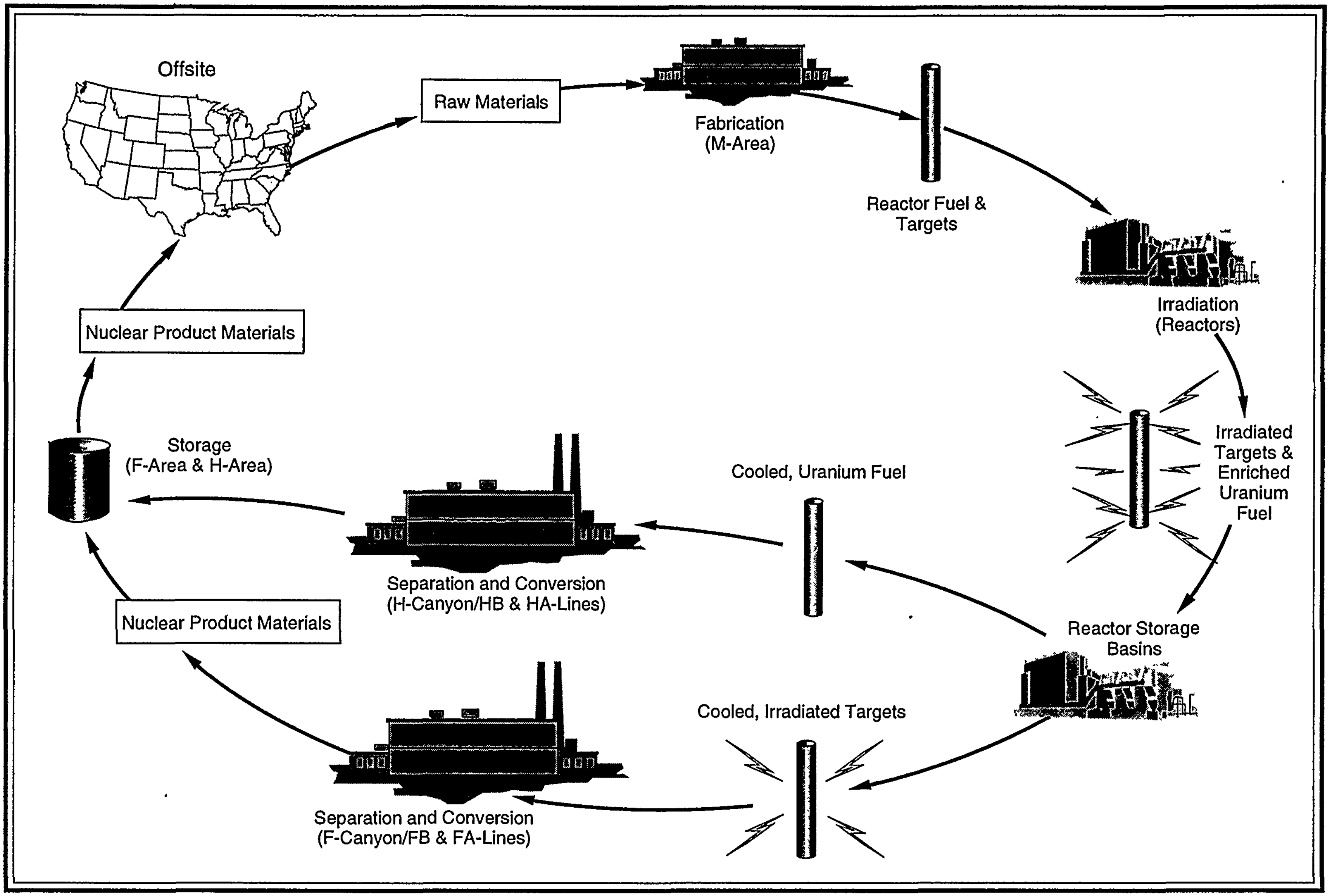

Figure 1-3. Historic nuclear materials production cycle at the Savannah River Site.

the materials produced at the SRS. Figure 1-4 shows the historic processing and recovery cycle for scrap materials received from off the SRS. Figure 1-5 shows the historic reprocessing cycle for spent fuel received.

In March 1992, DOE suspended chemical reprocessing and recovery activities at the SRS to address a potential safety concern regarding the capacity of the F- and H-Canyon ventilation systems to withstand an earthquake. Each canyon exhaust ventilation system functions to maintain the air pressure in the portions of the facilities where chemical separations activities occur at a low (i.e., negative) pressure with respect to other areas in the canyon building and the atmosphere. This ensures that air is always drawn into the canyon and helps prevent the spread of any radioactive material that might be inadvertently released inside the hot or warm canyon.

The negative pressure is maintained by using several centrifugal fans to draw the air out of the canyons. The air is pulled through a large high-efficiency particulate air filter (which removes more than 99 percent of all material with a diameter of 0.3 microns and larger) and passed across instruments that verify the filter is working properly and that no airborne radioactivity above applicable limits is released. Then the air is discharged to the atmosphere through a 61-meter (200-foot) stack behind the canyon. The concrete stack is lined with a ceramic material similar to the way in which a fireplace is lined with ceramic brick. This liner is essentially free-standing inside the stack. Previous engineering evaluations determined that the canyon ventilation systems would be able to withstand the impact of an earthquake that generated a lateral ground acceleration as high as 0.2*g*. Earthquakes beyond this magnitude would damage the ventilation system and the canyon to the extent that operations in the canyon would not be possible. At the SRS, earthquakes of this magnitude are likely to occur once every 5,000 years. L11-17 TC

In March 1992, a periodic design review determined that the stack liner could fail in a less severe earthquake, during which the stack would stay rigid and intact but the liner would sway, striking the inside of the stack and possibly causing the liner to break and collapse. In such an instance the ventilation system and the canyon systems would still be physically intact and operable but would not function properly because the debris from the stack liner would clog the air passage at the base of the stack.

To address such a possible situation, appropriate response procedures were prepared, personnel were trained, and response drills were conducted. All aspects of relevant facility operations were evaluated through DOE and contractor operational readiness reviews pursuant to the Unreviewed Safety

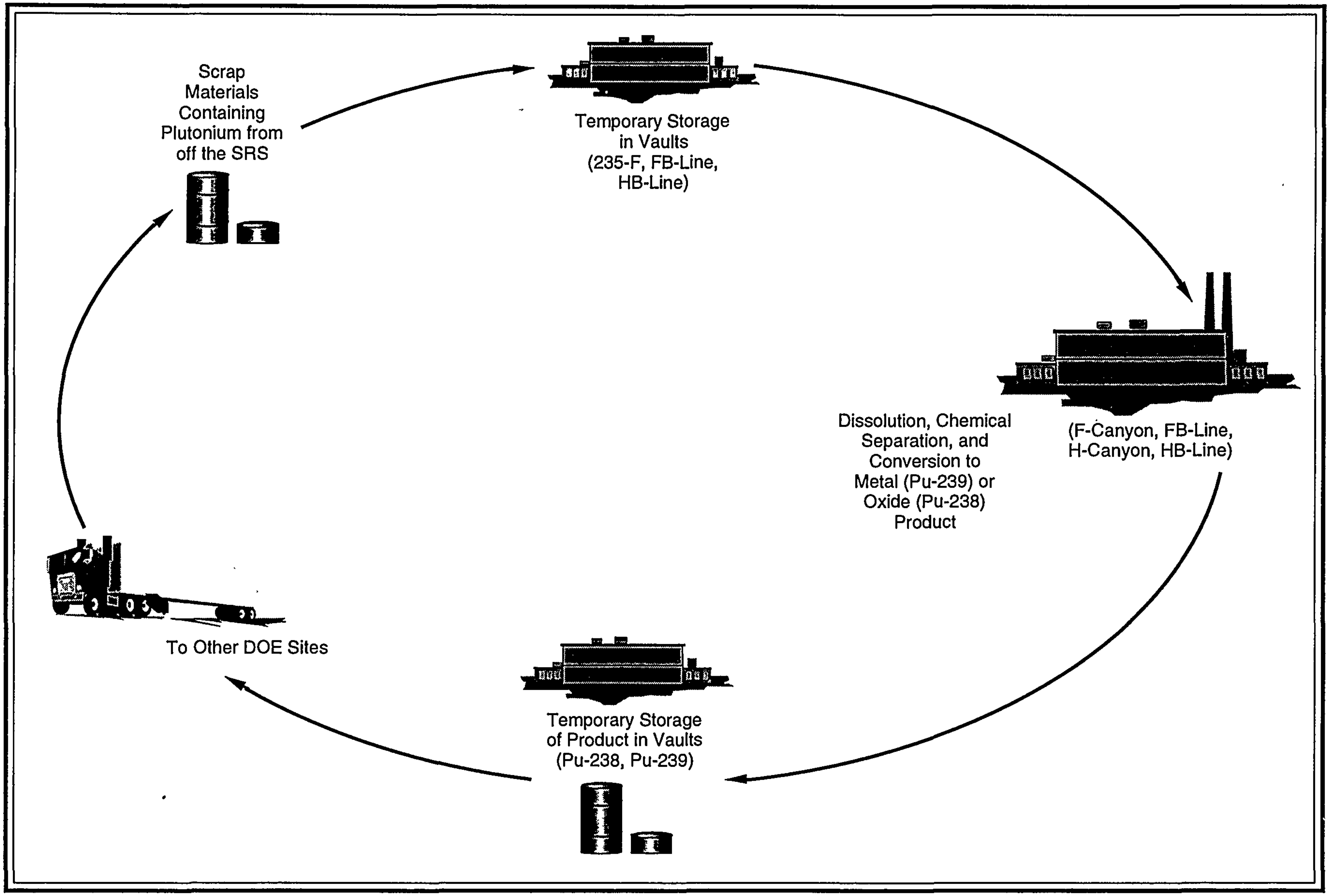

Figure 1-4. Historic scrap recovery cycle for plutonium at the Savannah River Site.

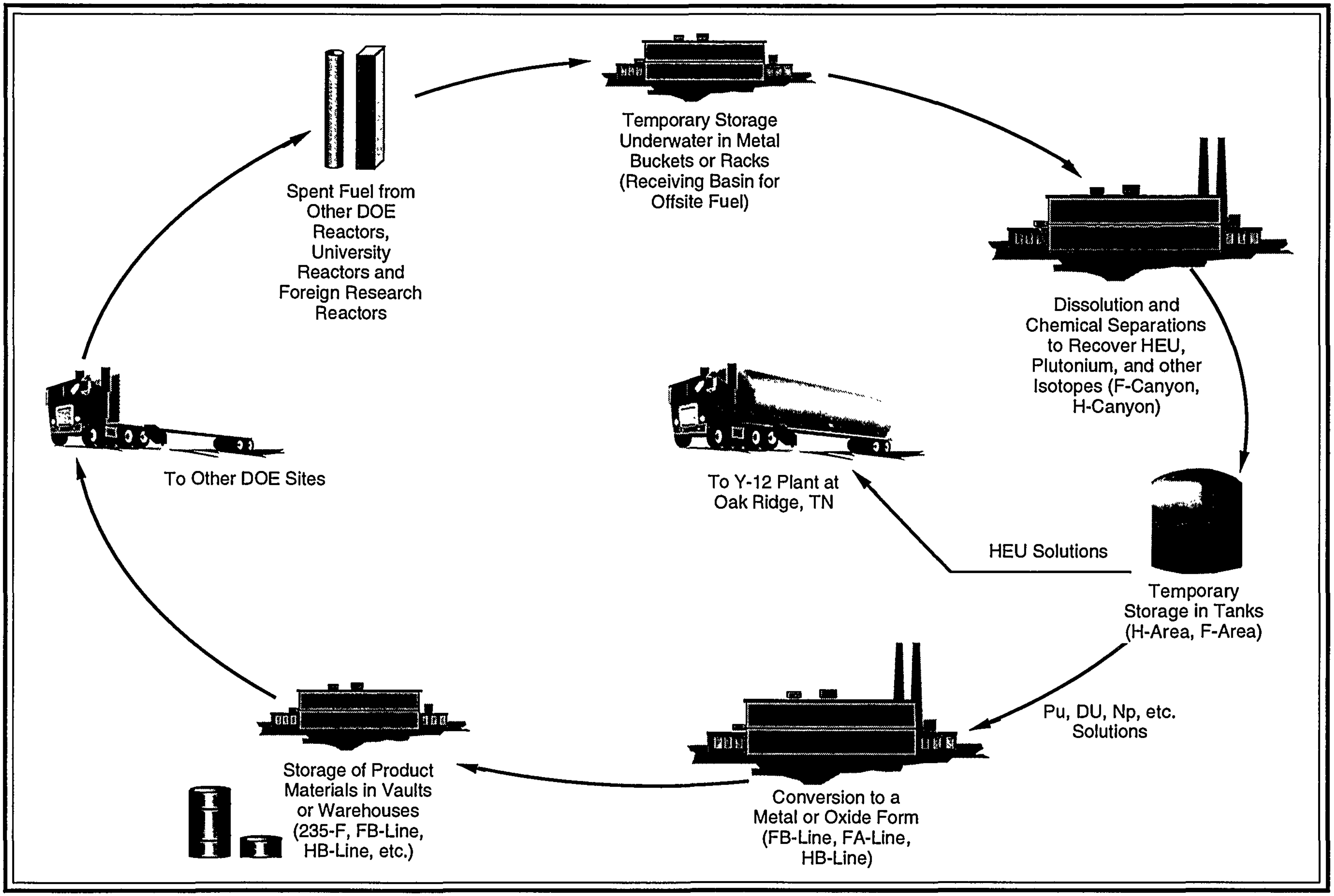

Figure 1-5. Historic reprocessing cycle for offsite spent fuel at the Savannah River Site.

Question program required by DOE Order 5480.21. At the conclusion of these activities, DOE concluded that the facilities were ready for safe resumption of processing operations.

However, in April 1992, before operation of the F- and H-Canyons could resume, the Secretary of Energy directed that the SRS phase out defense-related chemical separations activities in these facilities (DOE 1992a). World events in the late 1980s and early 1990s resulted in the end of the L11-17 TC Cold War and a reduction in the demand for new material for nuclear weapons. DOE stopped operating the SRS reactors to produce strategic isotopes. After the Secretarial decision in April 1992, DOE did not process nuclear materials at the SRS chemical separations facilities to recover special isotopes, with the exception of scrap materials containing plutonium-238. DOE continued the processing of plutonium-238 to support future National Aeronautics and Space Administration (NASA) exploratory space missions. In February 1995, DOE resumed F-Canyon operations to eliminate the risks involved in storing plutonium in solution form (DOE 1994d).

1.2 Purpose and Need for Action

The cessation in processing operations resulted in a large inventory of nuclear materials being caught in various stages of the historic production cycle (fabrication, irradiation, reprocessing, and recovery). These materials include irradiated and unirradiated reactor fuel, targets, and components; solutions containing dissolved nuclear materials and recovered isotopes in stainless-steel tanks; and product and scrap forms of metals or oxides in containers (cans, drums, etc.) typically used for temporary storage or shipment off the Site.

While DOE is evaluating various strategies for the long-term management and disposition of nuclear material (Section 1.6 describes these evaluations), the large inventory of nuclear materials at the SRS TC requires continued management. Some of the methods utilized to store these materials potentially pose risks to the environment or the safety and health of SRS workers or the public because, at the time DOE suspended the production cycle, many of these materials either were in a form or were stored in a manner that was intended only for a temporary period (e.g., 1 to 2 years). In some cases, the material's physical or chemical form poses a risk; in other cases, the material simply needs to be repackaged or moved to another location to ensure its safe storage. As the materials are currently stored, a number of accidents could result in the release of radioactivity to the environment and exposure of workers and the public. By taking action to alter the physical or chemical form of the materials or to alter the manner in which they are stored, DOE can reduce or eliminate the potential for their accidental release. Figure 1-6 shows how the implementation of DOE's proposed actions could ultimately reduce public risk due to accidents; Appendix E contains the data to support this figure.

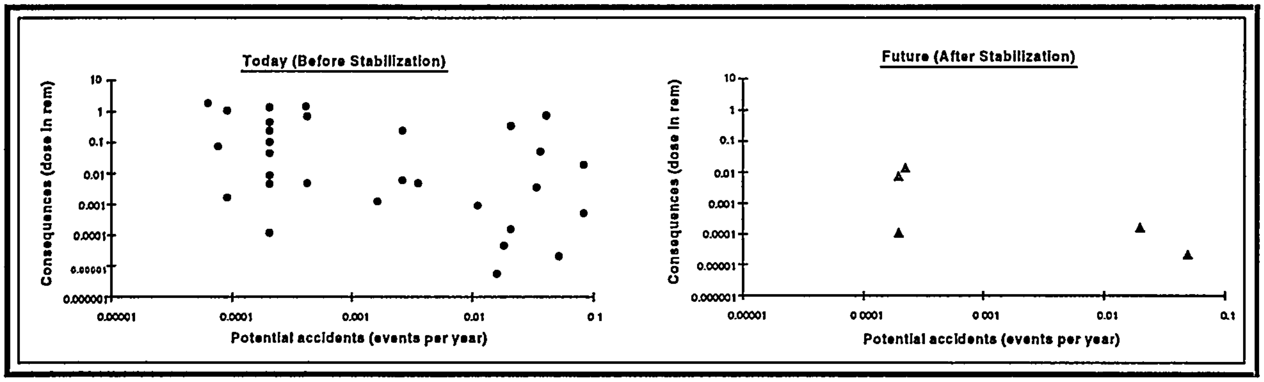

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Figure 1-6. Public risk profile before and after stabilization actions (assuming implementation of preferred alternatives).

While reducing risks to the public, workers, and the environment is the primary purpose and need for the proposed action, some nuclear materials stored at the SRS contain special isotopes that could be required to support ongoing DOE programs. Although the need for strategic isotopes has been greatly diminished by the end of the Cold War, certain materials require additional processing or conversion to forms suitable both for continued safe storage and potential use.

1.3 Defense Nuclear Facilities Safety Board Review

The Defense Nuclear Facilities Safety Board (DNFSB) is an independent organization established by Congress to provide oversight of DOE. On May 26, 1994, the DNFSB transmitted Recommendation 94-1 to the Secretary of Energy (DNFSB 1994). In its recommendation, the Board stated:

> The halt in production of nuclear weapons and materials to be used in nuclear weapons froze the manufacturing pipeline in a state that, for safety reasons, should not be allowed to persist unremediated. The Board has concluded from observations and discussions with others that imminent hazards could arise within two to three years unless certain problems are corrected.
>
> We are especially concerned about specific liquids and solids containing fissile materials and other radioactive substances in spent fuel storage pools, reactor basins, reprocessing canyons, processing lines, and various buildings once used for processing and weapons manufacture.
>
> It is not clear at this juncture how fissile materials produced for defense purposes will eventually be dealt with long term. What is clear is that the extant fissile materials and related materials require treatment on an accelerated basis to convert them to forms more suitable for safe interim storage.

The Board is especially concerned about the following situations:

- Several large tanks in the F-Canyon at the Savannah River Site contain tens of thousands of gallons of solutions of plutonium and trans-plutonium isotopes. The trans-plutonium solutions remain from californium-252 production; they include highly radioactive isotopes of americium and curium. These tanks, their appendages, and vital support systems are old, subject to deterioration, prone to leakage, and are not seismically qualified. If an earthquake or other accident were to breach the tanks, F-Canyon would become so contaminated that cleanup would be practically impossible. Containment of the radioactive material under such circumstances would be highly uncertain....

- The 603 Basin at the Idaho National Engineering Laboratory (INEL) contains deteriorating irradiated reactor fuel from a number of sources. This basin also contains sludge from corrosion of the reactor fuel. The seismic competence of the 603 Basin is not established.

- Processing canyons and reactor basins at the Savannah River Site contain large amounts of deteriorating irradiated reactor fuel stored under conditions similar to those at the 603 Basin at INEL.

- There are thousands of containers of plutonium-bearing liquids and solids at the Rocky Flats Plant, the Hanford Site, the Savannah River Site, and the Los Alamos National Laboratory. These materials were in the nuclear-weapons-manufacturing pipeline when manufacturing ended. Large quantities of plutonium solutions are stored in deteriorating tanks, piping, and plastic bottles.... Many of the containers of plutonium metal also contain plastic and, in some.... the plastic is believed to be in intimate contact with the plutonium. It is well known that plutonium in contact with plastic can cause formation of hydrogen gas and pyrophoric plutonium compounds leading to a high probability of plutonium fires....

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We are....following the plans for remedying several of the other situations listed. In general, these plans are at an early stage. In addition, we are aware of steps DOE has taken to assess spent fuel inventories and vulnerabilities. We also note that a number of environmental assessments are being conducted in relation to the situations we have listed above. Finally, we note that a draft DOE Standard has been prepared for methods to be used in safe storage of plutonium metal and plutonium oxide.

These actions notwithstanding, the Board is concerned about the slow pace of remediation. The Board believes that additional delays in stabilizing these materials will be accompanied by further deterioration of safety and unnecessary increased risks to workers and the public.

Therefore the Board recommends:

(1) That an integrated program plan be formulated on a high priority basis, to convert within two to three years the materials addressed in the specific recommendations below, to forms or conditions suitable for safe interim storage. This plan should recognize that remediation will require a systems engineering approach, involving integration of facilities and capabilities at a number of sites, and will require attention to limiting worker exposure and minimizing generation of additional waste and emission of effluents to the environment. The plan should include a provision that, within a reasonable period of time (such as eight years), all storage of plutonium metal and oxide should be in conformance with the draft DOE Standard on storage of plutonium now being made final....

(3) That preparations be expedited to process the dissolved plutonium and trans-plutonium isotopes in tanks in the F-Canyon at the Savannah River Site into forms safer for interim storage. The Board considers this problem to be especially urgent.

> (4) That preparations be expedited to repackage the plutonium metal that is in contact with, or in proximity to, plastic or to eliminate the associated existing hazard in any other way that is feasible and reliable. Storage of plutonium materials generated through this remediation process should be such that containers need not be opened again for additional treatment for a reasonably long time....
>
> (6) That preparations be expedited to process the deteriorating irradiated reactor fuel stored in basins at the Savannah River Site into a form suitable for safe interim storage until an option for ultimate disposition is selected....
>
> (8) That those facilities that may be needed for future handling and treatment of the materials in question be maintained in a usable state. Candidate facilities include, among others, the F- and H-Canyons and the FB- and HB-Lines at the Savannah River Site....

In response to the Board's recommendation, DOE developed an Implementation Plan (DOE 1995a) to address each concern in parallel with this EIS. The Plan contains detailed schedules and information on proposed DOE actions to alleviate the concerns raised by the DNFSB.

On August 3, 1995, the Chairman of the DNFSB transmitted a staff report to the Assistant Secretary of Energy for Environmental Management (DNFSB 1995). The DNFSB report identified the need for DOE to address leaking containers stored in the Receiving Basin for Offsite Fuel (RBOF), in addition to the other conditions identified in Recommendation 94-1. The DNFSB report stated:

> "...corroding spent fuel in the Receiving Basin for Offsite Fuel (RBOF) is releasing more than twice the amount of fission products into the basin water than the corroding Mark-31 targets are releasing into the L-Basin. The significant corrosion is contaminating the facility, generating significant waste, and contributing to personnel exposure." TC

In response to the DNFSB report, DOE has further evaluated the nuclear materials stored in the Receiving Basin for Offsite Fuel and has proposed actions to alleviate the Board's concerns with these materials, as well as those identified earlier in Recommendation 94-1. This EIS evaluates the potential environmental impacts from actions proposed by DOE in response to SRS-related concerns raised by the Board.

1.4 Categories of Nuclear Materials

For this EIS, DOE evaluated actions required to ensure the continued safe storage of the nuclear materials for the next 10 years. Ten years was used in response to public comments received during the scoping process for this EIS (DOE 1994l). The public felt that DOE might require that amount of time to analyze and make decisions for the eventual disposition of the materials, during which time

TC the materials must be safely stored. DOE has organized the inventory of nuclear materials at the SRS into three categories:

- Stable - These are materials that have physical and chemical forms that, combined with their storage configurations, do not currently pose an environmental, safety, or health concern and are not likely to pose a concern over the next 10 years.

- TE Candidates for Stabilization - DOE identified a number of environmental, safety, and health vulnerabilities associated with the storage of these materials; that is, conditions that might cause radiological exposure of workers or the public or contamination of the environment. The vulnerabilities might be due to the physical condition, chemical composition, or the manner in which the materials are stored (e.g., packaging or storage environment).

- L14-4 Programmatic - Materials that contain special isotopes that could be needed to support DOE programs. In their current forms, these materials are unsuitable for continued storage and are not usable. In many cases, the current forms of these materials pose the same vulnerabilities as the materials identified as Candidates for Stabilization (e.g., radioactive solutions).

This EIS analyzes the potential impacts that could be associated with the management of nuclear materials related to past production activities and missions of the SRS. All materials within the scope of this EIS exist in a specific physical form and, in accordance with DOE requirements, the SRS has maintained material control and accountability (MC&A) records on the amounts, forms, and physical storage locations of these materials. Actions proposed in this EIS would not create any new nuclear materials.

L2-1 The SRS maintains MC&A records for three types of nuclear material that are not within the scope of this EIS – tritium (and irradiated materials containing tritium), materials high in plutonium-238 content, and the plutonium-239 solutions in F-Canyon. DOE has addressed the recycling of existing inventories of tritium, an ongoing SRS program, in an environmental assessment (DOE 1986). In addition, DOE has addressed future tritium activities in the Draft Tritium Supply and Recycling Programmatic EIS (DOE 1995g). Similarly, the processing of plutonium-238 for NASA space missions (e.g., Cassini) was addressed in an environmental assessment (DOE 1991). Further, DOE has prepared a separate environmental assessment for future plutonium-238 processing operations that might be required (DOE 1995b). This EIS on the Interim Management of Nuclear Materials does, however, include a small amount of plutonium-238 contained in scrap from previous operations.

The plutonium solutions in F-Canyon, which are currently being stabilized, were the subject of a separate EIS (DOE 1994d).

The scope of this EIS does not include residual levels of nuclear materials contained in low-level, high-level, transuranic, and mixed types of radioactive waste. The SRS Waste Management EIS (DOE 1995c) evaluates the impacts from operations required to manage radioactive waste. There are residual levels of nuclear materials contained in production, processing, handling, or storage facilities scheduled for decontamination and decommissioning (D&D). These residual materials are also not included within the scope of this EIS. DOE will prepare separate NEPA documentation to evaluate impacts from D&D activities for such facilities, as appropriate. There are also residual levels of nuclear materials in radioactively contaminated areas of the SRS. These areas, such as seepage basins no longer used for low-level liquid waste, are the subject of environmental restoration programs at the SRS. These residual levels of nuclear materials are also not included in the scope of this EIS. L2-1

1.5 Categorization Methods

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1.5.1 STABLE MATERIALS AND CANDIDATES FOR STABILIZATION

DOE categorized Stable materials and Candidates for Stabilization as a result of several reviews. Within the past 2 years, DOE completed two nationwide reviews of how it stored nuclear materials at the SRS and other sites: TE

- *Spent Fuel Working Group Report on Inventory and Storage of the Department's Spent Nuclear Fuel and Other Reactor Irradiated Nuclear Materials and Their Environmental, Safety and Health Vulnerabilities* (November 1993) (DOE 1994a)

- *Plutonium Working Group Report on Environment, Safety and Health Vulnerabilities Associated with the Department's Plutonium Storage* (November 1994) (DOE 1994b)

The DOE Office of Environment, Safety and Health performed these reviews using teams of independent technical experts. Each report identified vulnerabilities associated with the continued storage of one or more nuclear materials at the SRS. The following sections summarize the scope of each review, the vulnerabilities identified with SRS materials, and the methods DOE used to categorize materials as Candidates for Stabilization or Stable.

1.5.1.1 Spent Fuel Working Group Report

The Department of Energy is storing nationwide large amounts of spent nuclear fuel and other reactor irradiated nuclear materials (RINM). In the past, DOE reprocessed these materials to recover plutonium and other isotopes. However, the Department has halted or is phasing out reprocessing operations. As a result, facilities designed, constructed, and operated to store materials for short periods of time have had to store materials for longer periods than originally intended. The extended use of the facilities, combined with the degradation of some facilities and that of their stored materials, has led to uncertainties about safety.

In August 1993, the Secretary of Energy assigned the Office of Environment, Safety and Health the responsibility to lead the DOE assessment of the environmental, safety, and health vulnerabilities associated with the storage of reactor irradiated nuclear material (DOE 1994a). DOE Operations Offices, laboratories, and Management and Operating Contractors designated personnel with the best technical knowledge of the inventory, data, operations, and safety bases for the storage facilities to participate in the assessment. Thirteen site teams gathered inventory and environment, safety and health information about their storage facilities, and completed questionnaires designed to collect information and data to characterize conditions and symptoms. From this information, the teams identified adverse conditions in terms of criticality, potential radioactive material release, potential radiation exposure, and institutional control failures. Appendix A discusses general vulnerabilities associated with RINM.

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The scope of this assessment was nationwide, involving 66 facilities at 11 sites where DOE stores reactor irradiated nuclear materials in basins, pools, canals, canyons, inactive reactors, warehouses, hot cells, vaults, wells, casks, and burial grounds. The assessment identified facilities that DOE should consider for priority attention, programmatic issues DOE should consider in its decisionmaking, and specific vulnerabilities at some of the facilities. It did not include fuel in active reactors, waste products, and irradiated structural materials. The assessment defined vulnerabilities in nuclear facilities as conditions or weaknesses that might lead to radiation exposure to the public, unnecessary or increased exposure to workers, or release of radioactive materials to the environment. Vulnerabilities are also caused by loss of institutional control, such as cessation of facility funding or reductions in facility maintenance and control. The assessment identified several areas as warranting priority attention to avoid increases in worker radiation exposure and cost during cleanup. At the SRS the assessment identified the K- and L-Reactor Disassembly Basins, and stated:

> Corrosion of fuel and target materials in the water basins and its effects constitute the major ES&H (Environment, Safety, and Health) vulnerability at the SRS pertaining to stored RINM. Corrosion is occurring in K- and L-Reactor basins and it is becoming increasingly difficult to maintain the cesium-137 activity within the administrative limit. Continued corrosion will eventually impact the physical integrity of stored materials. Such an eventuality would impact criticality, personnel radiation exposure, and fuel retrievability and disposal. The mechanisms and consequences of the corrosion are being addressed by WSRC (Westinghouse Savannah River Company) and the levels of contamination are low, however, fissile material such as uranium, plutonium are being released to the basin water which constitutes an ES&H vulnerability. Left unmitigated, the long term consequences of this situation could be severe.
>
> Based on the assessment conducted by the Working Group Assessment Team, the condition of the L-Reactor basin constitutes the greatest vulnerability as a consequence of the severity of the corrosion that is taking place, the quantity of stored material, and the level of the activity in the water. Next in degree of vulnerability is K-Reactor basin followed by P-Reactor basin, F-Canyon, H-Canyon, and RBOF (Receiving Basin for Offsite Fuel) in that order.

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The following paragraphs summarize the findings of the assessment team for the SRS facilities:

- L-Reactor Disassembly Basin - Delays and the subsequent suspension of reprocessing at the SRS have resulted in fuel and target residence times in the reactor basin significantly greater than those originally anticipated. Reactor basins were originally intended only for storage for approximately 12 to 18 months. The basin contains approximately 13,000 irradiated Mark-31 targets, 500 Mark-22 assemblies, and 600 other targets. The Mark-31 targets contain plutonium-239 in the uranium-238 matrix, the Mark-22 fuel contains uranium-235 in a uranium/aluminum alloy, and the other targets contain primarily cobalt-60. This material (and most other material in the reactor basins) has been stored for 5 years or longer.

  The targets and fuel are aluminum-clad. The Mark-31 targets (sometimes referred to as "slugs" due to their short cylindrical shape) are stored in stainless-steel buckets in the basin. The Mark-22 fuel and the other targets are stored either vertically on stainless-steel hangers or horizontally in slotted aluminum racks. The fuel suspended on hangers is corroding severely at the aluminum-to-stainless-steel interface region where a galvanic couple has formed. Relatively little corrosion (i.e., pitting or general) is occurring on cladding removed from the end region. However, corrosion is occurring in localized regions where the aluminum-oxide protective coating has been damaged; DOE assumes that cladding penetrations have occurred based on studies on representative nonirradiated alloys.

  The corrosion behavior observed on the Mark-31 targets stored in stainless-steel buckets is in sharp contrast to the behavior of the Mark-22 fuel. The Mark-31 targets have depleted

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uranium-238 as a core and are clad only with a thin layer of aluminum [nominal thickness of 0.076 centimeter (0.030 inch)]. The targets were designed with a thin layer of cladding to minimize waste generated during chemical dissolution and processing to recover the plutonium-239 produced through irradiation. A primary consideration in the cladding design was the expectation that the targets would be stored for only a short duration in the reactor disassembly and canyon basins (approximately 12 to 18 months), thus minimizing the potential for corrosion. Extensive pitting corrosion has penetrated the cladding, and corrosion of the uranium target material is releasing uranium, plutonium, and fission products to the basin water. DOE recently placed the buckets in stainless-steel boxes with lids to help confine the corrosion products. Continued corrosion will accelerate the transport of fissile materials into the water; subsequent material deposition and concentration in sludge and structural and water treatment components will increase concerns about possible criticality. Efforts are in process to remove this sludge by vacuuming, but corrosion will continue. The continued release of fission products to the basin and the subsequent cleanup will result in exposures to personnel.

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- <u>K-Reactor Disassembly Basin</u> - This basin contains approximately 900 Mark-16 and -22 fuel assemblies, 200 Mark-31 targets, and 200 other targets. The fuel and targets are stored in the same manner as those in the L-Reactor basin. The physical condition of the materials is deteriorating in the same way. The vulnerabilities applicable to the storage situation in the L-Reactor basin are applicable to the K-Reactor basin. The primary difference between the two basins is that the K-Reactor basin contains fewer Mark-31 targets, which are the materials that have exhibited the most extreme evidence of corrosion and physical deterioration. The vacuuming of sludge from the K-Reactor disassembly basin is under way.

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- <u>P-Reactor Disassembly Basin</u> - This basin contains approximately 500 Mark-22 fuel assemblies, 60 targets (slugs) used for the production of californium-252, and 2 Mark-42 assemblies used to produce plutonium-242. The fuel and targets are stored in the same manner as those in the L- and K-Reactor basins. The Mark-42 assemblies are stored in aluminum cans hung in a vertical position on stainless-steel hangers. "Although there is no evidence of corrosion on the surface of the fuel assemblies, the general corrosion of the components, including galvanic corrosion at the aluminum-stainless steel interfaces of the Mark-42 containers, aluminum tools, and the horizontal storage racks is judged to be the most severe in the P-Reactor basin" (DOE 1994a). The vulnerabilities applicable to the storage situation in the L- and K-Reactor basins are also applicable to the P-Reactor basin. The primary difference is that P-Reactor materials have been in storage a much shorter time than those in the L- and K-Reactor basins. P-Reactor basin contains the smallest amount of fuel and does not contain Mark-31 targets.

- F-Canyon Storage Basin - This basin contains approximately 2,500 Mark-31 targets (or slugs) stored in buckets. "If observed corrosion continues unmitigated, increased releases of fissile and radioactive materials are probable." The targets are "remaining in a non-favorable environment for far longer than that envisioned or anticipated." The "corrosion of the slugs and resultant nuclear material release would not significantly impact ES&H while the fuel (targets) remains in the F-Canyon; however, retrievability and handling would be encumbered" (DOE 1994a). In other words, if the material is left in this condition, it will become increasingly difficult to remove for long-term storage or disposition.

- H-Canyon Storage Basin - This basin contains 13 Mark-16 and Mark-22 fuel assemblies grouped in five bundles. No corrosion was detected at the time of the vulnerability assessment, but the similarity of the storage conditions to the reactor basins and F-Canyon basin make corrosion likely. As with the materials in the F-Canyon basin, continued storage will increase the difficulty of removing the material for long-term storage or disposition.

- Receiving Basin for Offsite Fuel - This basin contains approximately 3,000 irradiated fuel elements (assemblies, rods, tubes, cans, etc.). With the following limited exceptions, aluminum-clad fuels in storage and the aluminum racks that have been in the basin for more than 30 years appear to be in good condition. The RBOF contains fuel received from the decommissioned Taiwan Research Reactor (TRR). The TRR fuel was transported as part of the DOE Fuel Movement Program, which was a special project sponsored by the U.S. Department of State and the government of Taiwan to transport fuel from the decommissioned reactor to the SRS for storage and chemical processing. The project was independent of the previous DOE program and proposed policy for the receipt of foreign research reactor fuel containing uranium of U.S. origin (see DOE 1995e). The TRR fuel rods are natural uranium metal clad in aluminum. Before the fuel rods were loaded in Taiwan, each was inspected to identify any evidence of cladding failure. Eighty-one of the fuel rods were known to have breached or failed cladding when they were loaded for transport to the United States. In some cases the extent of cladding breach was severe, ranging from large gaping cracks along the length of the fuel rod to complete failure of the rod into disconnected segments. The failed fuel was placed in sealed aluminum canisters filled with an inert gas to minimize further corrosion and transported in spent fuel casks to the SRS in 1990 and 1991. Since then, approximately 16 of the aluminum canisters have been observed to be releasing gas. Based on this observation and TC basin monitoring reports, it is reasonable to conclude that radioactivity is being released to the basin water.

The RBOF also contains approximately 60 aluminum canisters of declad Experimental Breeder Reactor (EBR)-II slugs. The EBR-II slugs were originally depleted uranium metal clad in stainless steel. The stainless-steel cladding was removed and each depleted uranium slug was placed in an aluminum canister filled with an inert gas. The declad slugs were transported to the SRS in 1986 and have been stored in their aluminum canisters in RBOF since then. A single canister of the EBR-II slugs has been observed to be releasing gas, and it is reasonable to conclude that this canister is releasing radioactivity to the basin water.

Finally, the RBOF contains 40 Mark-31 targets stored in metal buckets on the bottom of the basin. At least six of the Mark-31 targets were known to have failed cladding before being stored in RBOF. The six targets with failed cladding are in aluminum canisters.

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Based on the extent of the vulnerabilities identified, DOE categorized the materials in the L-, K-, and P-Reactor Disassembly Basins as Candidates for Stabilization. DOE categorized the fuel and target materials in the F- and H-Canyon storage basins as Candidates for Stabilization, primarily because similar vulnerabilities are likely to develop for these materials during the next 10 years. DOE has also categorized the material stored in the Receiving Basin for Offsite Fuel that has failed cladding or that are stored in failed canisters as Candidates for Stabilization. Specifically, there are 81 canisters of failed Taiwan Research Reactor fuel, 1 leaking canister of Experimental Breeder Reactor-II slugs, and 6 failed Mark-31 targets. Based on studies performed at the SRS, DOE expects minimal corrosion of the intact aluminum fuels stored in RBOF given the high quality of the water and the physical condition of the facility and equipment. Therefore, DOE has categorized the intact fuel and targets stored in RBOF as Stable. DOE believes the primary source of radioactivity being released to the basin water in RBOF is the limited number of fuel rods or targets that have failed cladding or that are stored in failed canisters. Removal of these materials from the basin is required to stop the release of soluble fission products into the basin and to minimize any further radioactive releases to the basin water.

If, after removing the RBOF materials described above [i.e., the limited amount of Taiwan Research Reactor fuel, Experimental Breeder Reactor-II slugs, and all Mark-31 targets (see Section 2.7.8)], DOE determines that additional fuel, targets, or canisters have failed (e.g., as indicated by additional gas releases from a canister, or visible failure of cladding or canisters), the Department would categorize those materials as Candidates for Stabilization. DOE would then perform the appropriate

National Environmental Policy Act review and evaluation for the stabilization of any additional materials in RBOF that are determined at a later date to be failed (e.g., a Supplement Analysis).

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1.5.1.2 Plutonium Working Group Report

Approximately 26 metric tons (28.7 tons) of plutonium, not including the amounts in assembled and disassembled nuclear weapons, are located throughout the nuclear weapons complex. In March 1994, the Secretary of Energy commissioned a comprehensive assessment to identify and prioritize the environmental, safety, and health vulnerabilities that arise from the storage of plutonium in DOE facilities and to determine the most dangerous and urgent of these vulnerabilities (DOE 1994b). These vulnerabilities include the degradation of plutonium materials and packaging and weaknesses in facilities and administrative controls, which can result in inadvertent releases of plutonium to the environment.

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The Plutonium Working Group combined the expertise of DOE staff members, Management and Operating contractors, consultants, and 45 stakeholder groups. The working group teams required 2 to 3 weeks at each site in addition to weeks of preparation and report writing. The basis for the assessment methodology used by the group is known as Target-Barrier-Hazard analysis. In general, this approach considers a material to be a "hazard" and then evaluates "barriers" that separate the workers, the public, or the environment from the hazard. The assessment teams gathered and evaluated site information. Facility inspections examined barriers and considered potential challenges to barriers at individual facilities. The team also collected information about plutonium types and inventories during inspections. The teams identified weaknesses, and the team leaders categorized these weaknesses as material/packaging, facility condition, or institutional vulnerabilities. These vulnerabilities were then classified on the basis of the likelihood of potential accidents or events due to the vulnerabilities of the worker, public, or environment. Appendix A discusses general vulnerabilities of plutonium materials and storage facilities.

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Specifically, the working group report evaluated the storage of nearly all the plutonium that is not in intact nuclear weapons. It reviewed plutonium forms and packaging with the exception of residual plutonium from underground nuclear tests; plutonium in low-level, high-level, and transuranic wastes; and plutonium in very low residual levels in facilities undergoing decontamination and decommissioning. (DOE evaluated plutonium in spent fuel and irradiated targets in the spent fuel study described in Section 1.4.1.1.) This assessment included transuranic elements such as neptunium, americium, curium, and californium. It identified 299 environmental, safety, and health

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vulnerabilities at 13 sites. The following paragraphs summarize the vulnerabilities that involve SRS materials.

Solution Vulnerabilities. The Plutonium Working Group Vulnerability Report (DOE 1994b) presents the following information on solution vulnerabilities:

> F-Canyon has 320,000 liters of plutonium solution and 14,000 liters of americium and curium in solution in 15 tanks. H-Canyon has 50,000 liters of plutonium solution (about 13,000 liters containing plutonium-242 and the remainder containing both weapons grade and fuel grade plutonium) and 6,000 liters of neptunium solution in four tanks. These solutions are unstable and corrosive and might breach containers, resulting in releases of radioactive materials. Such releases could cause exposure of workers and the public and environmental contamination. Unanticipated high local plutonium concentrations in these tanks could also lead to criticality accidents. A recent study by DOE's Office of Environment, Safety and Health on F-Canyon plutonium solution storage (DOE/EH-0397P) determined that the potential for inadvertent criticality could be significant. Individual F- and H-Canyon tanks hold from 2,000 to nearly 55,000 liters of solutions. These tanks require continuous monitoring for corrosion, sampling for adjustment of solution chemistry and periodic reagent additions to maintain liquid levels and prevent formation of solids. In 1993, plutonium precipitation in two tanks in F-Canyon prompted the immediate addition of boric acid for criticality control. Following the boric acid addition, mixing was begun and nitric acid was added to redissolve the plutonium. These tanks contain plutonium solutions awaiting decisions on further processing. Continued storage of these highly dispersible solutions creates significant vulnerabilities to workers and the environment.
>
> L11-14 TC The tank containing 14,000 liters of americium and curium solution is the largest single source of radioactivity (approximately 220,000 curies) in F-Canyon. This solution has been in storage since 1983, and tank corrosion is a concern. Recently, the tank was disconnected from the cooling water system to prevent the possibility of a leak causing large off-site releases to the environment and exposure of the public. The solution is self-heating and remains at a temperature slightly less than 60°C, which causes a high rate of evaporation. Adjustments for solution chemistry and volume must be made frequently. Tank contents are susceptible to spills and leaks and could be widely dispersed by a major facility accident.

Due to the vulnerabilities identified, DOE categorized these solutions as Candidates for Stabilization.

Metal, Oxide, and Scrap/Residue Vulnerabilities. The Vulnerability Report (DOE 1994b) presents the following information:

> FB-Line and Building 235-F have more than 400 packages of plutonium metal and metal alloys and about 2,400 packages of plutonium oxides and compounds. Material/packaging properties that could lead to worker exposure are: reactive and/or corrosive compounds; plastics which degrade due to radiolytic and thermal decomposition (80 percent of the packages contain plastic); metal subject to oxidation and subsequent expansion due to oxide formation; and unknown and uncharacterized materials and packaging (i.e., the chemical composition is not completely known). The more than 2,800 packages identified above contain combinations or mixtures of the following materials:
>
> - Plutonium-uranium oxides (including normal and enriched uranium), oxides mixed with transuranics including neptunium and americium, and scrap/residues such as plutonium alloys are present in over 500 packages that have not been fully characterized and have unknown packaging. This could lead to unsuspected reactions between materials and an eventual breach of packaging.

- Fuel-grade plutonium (a higher specific activity material containing up to 18 percent plutonium-240 in addition to plutonium-239), is present in about 600 packages. This plutonium generates heat, thereby accelerating degradation of plastics and increasing the chances of packaging failure.

- Scrap/residues received from other DOE sites in over 150 different forms, including incinerator ash, graphite, chloride-bearing residues and partly characterized and potentially reactive compounds such as plutonium nitride, are present in over 600 packages. These include most of the packages of oxides and scrap/residues and packages of fuel grade plutonium mentioned above.

- Scrap/residues from plutonium metal production present in 700 of the more than 2,800 packages, contain calcium metal and corrosive fluoride compounds that can react with moisture and air and undergo radiolysis.

In Building-235F, many containers have been stored for as long as 20 years in their original shipping containers, with no provisions for inspection or monitoring.

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The ultimate goal of the plutonium vulnerability assessment was to facilitate the safe and stable interim storage of plutonium materials held by the Department. Based on the assessment, DOE concluded that the plutonium package failures and facility degradation will increase in the future unless it addresses the problems in an aggressive manner.

Due to the vulnerabilities noted, DOE categorized the materials listed above as Candidates for Stabilization. [Due to the urgency of concerns related to the 300,000 liters of plutonium solutions stored in F-Canyon, DOE prepared a separate EIS evaluating alternatives for that material (see Section 1.6). DOE issued a Record of Decision in February 1995 to convert the F-Canyon plutonium solutions into a metal form for safe interim storage.]

1.5.1.3 Materials Not Included in the Spent Fuel and Plutonium Working Group Reviews

The scope of the Spent Fuel and Plutonium Working Group Reviews did not encompass all nuclear materials stored at the SRS. For each material not previously evaluated by an independent review, DOE performed an assessment to determine if the material poses an environmental, safety, and health concern or could pose a concern over the next 10 years. The assessment was performed by technical personnel responsible for the management of the nuclear materials in their current storage locations. Independent technical experts reviewed the results of the assessment, which consisted of a series of questions to evaluate qualitatively the inherent physical stability of the material, the current and projected physical condition of its storage container, and the potential for release of the material to the environment (DOE 1994).

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Of the materials not included in the Spent Fuel and Plutonium Working Group Reviews, only one was judged to pose an existing or potential concern. The SRS has approximately 228,000 liters (60,000 gallons) of highly enriched uranium solutions stored in stainless-steel tanks inside and outside H-Canyon. These solutions were encompassed by Defense Nuclear Facilities Safety Board Recommendation 94-1 (see Section 1.3). DOE has, therefore, included the highly enriched uranium solutions as candidates for stabilization. The solutions pose a criticality hazard that is inherently difficult to control for an extended period of time (i.e., until long-term storage or dispositions can be made). L11-14 TC

Although approximately 300,000 liters (78,000 gallons) of depleted uranium solutions are stored in stainless-steel tanks inside and outside F-Canyon and in the TNX Area, DOE categorized these materials as Stable. DOE did not consider these solutions to pose an environmental, safety, or health concern because they contain only trace quantities of fissile isotopes (uranium-235, plutonium-239, etc.) and represent a very low radiological hazard in comparison to that posed by other nuclear materials (see Appendix A). TC DOE categorized as Stable all other nuclear materials within the scope of this EIS that are stored at the SRS; this included a wide variety of nuclear materials containing special isotopes used to support sitewide operations, such as laboratory samples used in experimental work and encapsulated sources used for testing and calibrating equipment. TE

1.5.2 PROGRAMMATIC MATERIALS

DOE categorized certain nuclear materials as Programmatic after consultations with national laboratories and other appropriate Federal agencies (e.g., NASA). These consultations identified plutonium-242, neptunium-237, americium, and curium (various isotopes) as potentially required to support DOE programs and responsibilities.

L11-14 TC However, as discussed in Section 1.4, the bulk of these materials are in the form of solutions and would otherwise be considered candidates for stabilization. In other words, some action would still be required by DOE to ensure safe storage until decisions on long-term use or disposition could be made. The safety considerations are the primary reason for DOE's proposed actions. The potential programmatic use of the material is an important, but secondary, consideration.

At present, DOE uses plutonium-242 for research. In accordance with the Atomic Energy Act, specific details on the use of plutonium-242 are classified and restricted from unauthorized TC disclosure for the protection of national security. The SRS has plutonium-242 solution stored in a stainless-steel tank in H-Canyon that requires processing and conversion to a form suitable for safe storage and potential use. TC

Neptunium is a target material irradiated in a nuclear reactor to produce plutonium-238. Plutonium-238 is a thermal power source for remote terrestrial and space applications where solar collectors or chemical batteries are not feasible. The SRS has the remaining domestic inventory of recovered neptunium-237, the bulk of which is in solutions stored in stainless-steel tanks in H-Canyon. These solutions contain neptunium-237 that was recovered from the processing of irradiated highly enriched uranium fuels. In addition, the Site has a limited number of targets containing neptunium-237 that were designed for irradiation in the SRS reactors; with the shutdown of the reactors, these targets are no longer usable. To support the future production of plutonium-238, DOE must convert these materials to a form that it can store safely and potentially use later to fabricate new targets.

The approximately 14,000 liters (3,700 gallons) of solution stored in a single stainless-steel tank in F-Canyon represent a unique stockpile of americium and curium that DOE might need to support domestic and international research programs. The solution contains about 9 kilograms (20 pounds) of americium-243 and about 2.5 kilograms (5.5 pounds) of curium-244. The solution also contains iron, nickel, chromium, and other corrosion products from the stainless-steel tank. To be usable, sufficiently pure americium and curium must be extracted from the solution. In addition, approximately 60 small metal slugs are stored in the P-Reactor disassembly basin and 114 metal slugs and 65 metal target assemblies are stored in the Receiving Basin for Offsite Fuel. These slugs and target assemblies contain varying valuable amounts of the americium and curium isotopes. The 65 targets contain about 790 grams (1.7 pounds) of curium isotopes 244 to 248, primarily curium-246. There is essentially no americium in the targets, which are 1.2-meter (4-foot)-long tubes with a 10-centimeter (4-inch) diameter. The metal slugs contain significant amounts of heavy nuclides, including some americium-243 (Lowe 1994). L14-8 TC

DOE uses americium and curium isotopes in the production of californium-252, which is a neutron source for radiography and for nuclear medicine in the treatment of certain types of cancer, and for research in basic chemistry, nuclear physics, and solid-state chemistry. These isotopes are invaluable as feedstock materials for producing heavier transplutonium elements. Current feedstocks at the DOE Oak Ridge Reservation will last only about 8 more years. The SRS transuranium element stocks are the only remaining feedstock materials, and they probably will never be produced in such quantities again (Seaborg 1994). These forms require processing and conversion to produce a physical form that DOE can store safely and potentially use later.

Table 1-1 summarizes the inventory of nuclear materials at the SRS in the Stable, Programmatic, and TE Candidate for Stabilization categories of material. Appendix A contains a more detailed listing.

Table 1-1. SRS nuclear materials.[a]

	Description	Quantity	Location(s)
	Stable		
	Spent fuel	3,000 items	Receiving Basin for Offsite Fuel
	Unirradiated fuel, targets, reactor components, and scrap from fabrication operations	315,000 items	Buildings 305A, 313-M, 315-M, 320-M, 321-M, 322-M, and 341-M, K- and L-Reactor Assembly Areas
	Unirradiated fuel, targets, and reactor components	6,900 items	K- and L-Reactors
	Unirradiated and irradiated reactor components and control rods	420 items	C-, K-, L-, and P-Reactors
	Depleted uranium oxide	36,000 drums	R-Reactor, Buildings 221-1F, 221-12F, 221-21F, 221-22F, 707-R, 714-7N, 728-F, 730-F, and 772-7B
	Depleted uranium solutions	300,000 liters (78,000 gallons)	F-Canyon, F-Area Outside Facilities, and TNX
	Sources, standards, and samples	20,000 items	Sitewide
	Laboratory materials used in research and development	260 items	Savannah River Technology Center
	Programmatic		
	Plutonium-242 solutions	13,000 liters (3,500 gallons)	H-Canyon
L9-2 L14-4	Americium and curium solutions and targets	14,000 liters (3,800 gallons)	F-Canyon
		65 assemblies	Receiving Basin for Offsite Fuel
		60 slugs	P-Reactor disassembly basin
		114 slugs	Receiving Basin for Offsite Fuel
	Neptunium solutions and targets	6,100 liters (1,600 gallons)	H-Canyon
		9 targets	Building 321-M
	Candidates for Stabilization		
	Plutonium-239 solutions	34,000 liters (9,000 gallons)	H-Canyon
TE	Highly enriched uranium solutions	228,000 liters (60,000 gallons)	H-Canyon and H-Area Outside Facilities
	Plutonium vault materials	2,800 packages	FB-Line, HB-Line, Building 772-F, Building 235-F, and SRTC
	Mark-31 targets	16,000 slugs	K-Reactor, L-Reactor, F-Canyon, and RBOF
	Mark-16 and Mark-22 fuels	1,900 assemblies	K-, L-, and P-Reactors and H-Canyon
	Other aluminum-clad targets	1,800 slugs and assemblies	K-, L-, and P-Reactors
	Failed TRR[b] fuel and EBR-II[c] slugs	82 canisters	Receiving Basin for Offsite Fuel

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a. Appendix A contains a more comprehensive listing and description of these materials. Quantities of materials shown here are approximate. Quantities of radioactive solutions stored in tanks fluctuate due to natural evaporation and the addition of materials (e.g., nitric acid) to maintain chemistry within established parameters. Therefore, quantities listed in this table are approximate and might vary from quantities cited in previous DOE reports or studies.

b. Taiwan Research Reactor - 81 canisters.

c. Experimental Breeder Reactor-II - 1 canister.

Figure 1-7 shows the relative mass of nuclear material in each category. As the figure reflects, the vast majority (more than 98 percent) of the stored mass of nuclear materials falls within the Stable category. The high percentage of stable material is heavily influenced by the fact that much of the material in the stable category is depleted uranium stored in approximately 36,000 drums and approximately 315,000 miscellaneous items left from the fabrication process for SRS reactor components (fuel, targets, etc.), which contain varying amounts of uranium.

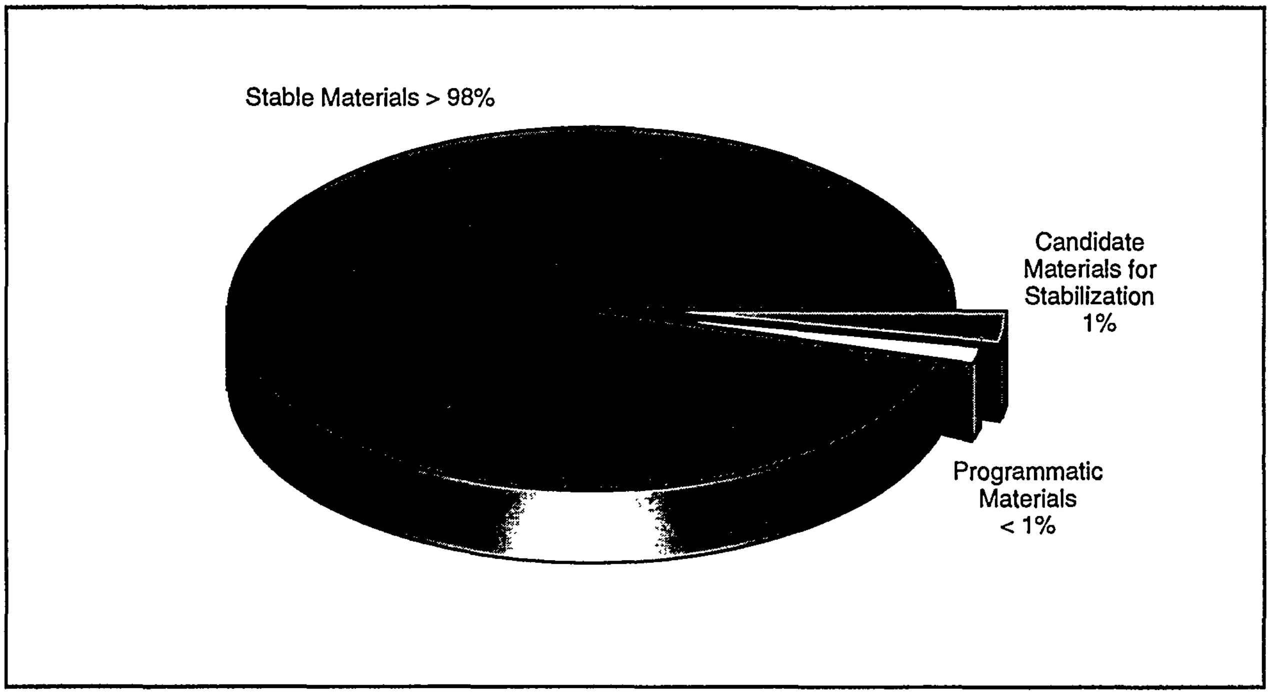

Figure 1-7. Amount of nuclear material in each category.

1.6 Relevant National Environmental Policy Act Documents

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F-CANYON PLUTONIUM SOLUTIONS ENVIRONMENTAL IMPACT STATEMENT

On March 17, 1994, DOE published (59 FR 12588) its intention to prepare the Interim Management of Nuclear Materials EIS to assess the interim management of nuclear materials stored at the SRS. The original scope of this EIS included the plutonium solutions stored in the F-Canyon facility. In May 1994 the Manager of the Savannah River Operations Office recommended that the Assistant Secretary for Defense Programs seek alternative arrangements for compliance with the National Environmental Policy Act (NEPA) to allow stabilization of the plutonium solutions in F-Canyon and the Mark-31 targets stored in the L-Reactor Disassembly Basin. The recommendation was based on the determination that the material presents risks to workers, the public, and the environment in the form of radiation exposure from normal operations and potential accidents, which DOE could reduce by converting the material to a solid stable form. In June 1994 the DOE Office of Environment, TE | Safety and Health performed an independent evaluation of the SRS request (DOE 1994e). That report characterized the following potential facility accidents to be of serious concern: (1) the potential for inadvertent criticality due to precipitation of plutonium from the F-Canyon solutions, and (2) potential radiological releases to the environment due to leakage of plutonium solutions through vessel cooling coils. The report did not conclude that the Mark-31 targets would be a serious concern over the following 12 to 20 months. In light of this evaluation, DOE determined that the appropriate action would be to prepare a separate expedited EIS to evaluate management alternatives for the F-Canyon plutonium solutions. On August 23, 1994, DOE published in the TC | *Federal Register* the notice of an amendment to announce the preparation of a separate EIS on these solutions. The Final EIS on F-Canyon Plutonium Solutions (DOE 1994d) became available on December 30, 1994. The Record of Decision was signed on February 1, 1995. The F-Canyon TE | Plutonium Solutions EIS is relevant to the assessment of cumulative impacts that could occur at the SRS during the period examined by this Interim Management of Nuclear Materials EIS (see Chapter 5).

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PROGRAMMATIC ENVIRONMENTAL IMPACT STATEMENT FOR STORAGE AND DISPOSITION OF WEAPONS-USABLE FISSILE MATERIALS

As announced in the *Federal Register* on June 21, 1994 (59 FR 31985), DOE is preparing this Programmatic EIS to evaluate the long-term storage of weapons-usable fissile materials, primarily plutonium-239 and highly enriched uranium, and the disposition of such materials that the President has declared surplus to national defense needs. As described above, the SRS has a large inventory of plutonium-239, highly enriched uranium, and other weapons-usable fissile materials that DOE will TE | include in the scope of the Programmatic EIS. The Programmatic EIS is, therefore, relevant because it evaluates alternatives for some of the materials discussed in this EIS. However, the implementation of decisions resulting from the Programmatic EIS could require several years to complete. Therefore, interim decisions on stabilization and storage alternatives for weapons-usable fissile materials are necessary until DOE can reach and implement those long-term decisions.

DISPOSITION OF SURPLUS HIGHLY ENRICHED URANIUM ENVIRONMENTAL IMPACT STATEMENT

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On April 5, 1995, DOE amended the scope of the Storage and Disposition of Weapons-Usable Fissile Materials Programmatic EIS by announcing in the *Federal Register* its intention to address the disposition of highly enriched uranium in a separate EIS (60 FR 17344). DOE based this action on the need to "move forward on a rapid path for neutralizing the proliferation threat of surplus HEU and to demonstrate to other nations the United States' nonproliferation commitment. The disposition of HEU will involve different time frames, technologies, facilities, and personnel than those required for the disposition of plutonium. Therefore, the decisions on surplus HEU disposition do not affect or preclude other decisions to be made on the long-term storage and disposition of other weapons-usable fissile materials, can proceed regardless of decisions pursuant to the PEIS, and are independently justified." The scope of the PEIS will continue to include the long-term storage of weapons-usable fissile materials, including HEU, and the disposition of surplus plutonium. DOE plans to complete the EIS on the disposition of surplus highly enriched uranium in early 1996.

The EIS on the disposition of surplus highly enriched uranium is related to this Interim Management of Nuclear Materials EIS in two ways: First, this EIS includes surplus highly enriched uranium materials at the SRS, which range from radioactive solutions stored in tanks to obsolete unirradiated reactor components (fuel assemblies) stored in concrete and metal vaults (see Appendix A). Under the alternatives evaluated in the EIS on the disposition of highly enriched uranium, DOE would blend existing inventories of highly enriched uranium with low enriched, natural, or depleted uranium to reduce the enrichment (i.e., the amount of uranium-235) and thus to eliminate its weapons usability. Similarly, this EIS evaluates the Blending Down to Low Enriched Uranium Alternative for stabilization of solutions and materials stored in reactor basins (see Chapter 2). Second, to implement blend-down alternatives for the disposition of highly enriched uranium, DOE is evaluating the potential use of the SRS chemical processing facilities (F- and H-Canyons) as one of four potential DOE and commercial blending sites. Alternatives involving the use of the SRS for the disposition of highly enriched uranium by blending down could occur in the same timeframe as the stabilization actions described in this EIS (i.e., within the next 10 years). In consideration of this, Chapter 5 of this EIS contains the cumulative impacts from potential operation of the canyons for disposition of highly enriched uranium. TC

ENVIRONMENTAL ASSESSMENT FOR THE PROPOSED INTERIM STORAGE OF ENRICHED URANIUM ABOVE THE MAXIMUM HISTORICAL STORAGE LEVEL AT THE Y-12 PLANT

The SRS has a large inventory of nuclear materials containing highly enriched uranium that could be consolidated for interim storage at the Y-12 Plant in Oak Ridge, Tennessee. These materials include a large portion of the approximately 315,000 items that remain from the fabrication of new (unirradiated) fuel for SRS reactors, approximately 228,000 liters (60,200 gallons) of highly

enriched uranium solutions stored in stainless-steel tanks in H-Area, and irradiated fuel from both
TE SRS and offsite reactors. Recent SRS operations consolidated the unirradiated fuel and leftover
materials that contain highly enriched uranium into forms suitable for transport and storage at the
Y-12 Plant. The conversion of the highly enriched uranium solutions into a highly enriched uranium
oxide is one of the management alternatives evaluated in this EIS, as is the dissolution and
reprocessing of irradiated SRS reactor fuel to recover highly enriched uranium. The Environmental
Assessment on Uranium Storage at the Y-12 Facility (DOE 1994f) includes the transport and storage
of SRS highly enriched uranium materials. Therefore, the Y-12 Environmental Assessment is
TE relevant to this EIS. DOE published this Environmental Assessment and its Finding of No Significant
Impact on September 14, 1995.

SAVANNAH RIVER SITE WASTE MANAGEMENT ENVIRONMENTAL IMPACT STATEMENT

On April 6, 1994, DOE issued a Notice of Intent in the *Federal Register* (59 FR 16194) to prepare an
SRS Waste Management EIS, to provide a basis for selecting a sitewide strategic approach to
managing present and future wastes generated at the Site. These wastes would be generated by
several activities including ongoing operations and potential actions, new missions, environmental
TC restoration, and decontamination and decommissioning programs. The Final SRS Waste
Management EIS (DOE 1995c), which became available on July 28, 1995, includes the treatment of
wastewater discharges in the Effluent Treatment Facility, F- and H-Area tank operations and waste
removal, and construction and operation of a replacement high-level waste evaporator in the H-Area
tank farm. In addition, it evaluates the Consolidated Incineration Facility technology for the
treatment of mixed waste. All the alternatives evaluated in this Interim Management of Nuclear
Materials EIS will result in the generation of waste (high-level, transuranic, mixed, etc.). Thus, the
TE SRS Waste Management EIS is relevant to this EIS because it evaluates management alternatives for
various types of waste that actions proposed in this EIS could generate. The SRS Waste Management
EIS is also relevant in the assessment of cumulative impacts that could occur at the SRS during the
TE period examined by this EIS (see Chapter 5). DOE issued the Record of Decision for the SRS Waste
Management EIS on September 22, 1995.

DEFENSE WASTE PROCESSING FACILITY SUPPLEMENTAL ENVIRONMENTAL IMPACT STATEMENT TE

On April 6, 1994, DOE issued a Notice of Intent in the *Federal Register* (59 FR 16499) to prepare a Supplemental EIS on the Defense Waste Processing Facility (DWPF) to examine the impacts of completing construction and operating the DWPF at the SRS. This supplement to an EIS that DOE issued in 1982 assisted the Department in deciding whether and how to proceed with the DWPF in light of changes to processes and facilities that had occurred since the issuance of the 1982 EIS. The Final EIS (DOE 1994g) was issued on November 25, 1994. The Record of Decision was signed on March 28, 1995 (60 FR 10589, April 12, 1995). The decision is to complete the construction and startup testing of the DWPF, and to operate the facility using the in-tank precipitation process after TC the satisfactory completion of startup tests.

One of the alternatives considered for the stabilization of materials in this Interim Management of Nuclear Materials EIS is vitrification using the Defense Waste Processing Facility. All the alternatives evaluated in this EIS would result in the generation of radioactive waste that DOE would have to handle or treat at facilities described in the SRS Waste Management EIS and the DWPF Supplemental EIS. Appendix D describes the estimated amounts of generated waste. The DWPF Supplemental EIS is also relevant in the assessment of cumulative impacts that could occur at the SRS during the period examined by this EIS. These impacts have been included in the cumulative impact evaluation discussed in Chapter 5.

PROGRAMMATIC SPENT NUCLEAR FUEL MANAGEMENT AND IDAHO NATIONAL ENGINEERING LABORATORY ENVIRONMENTAL RESTORATION AND WASTE MANAGEMENT PROGRAMS ENVIRONMENTAL IMPACT STATEMENT (SNF AND INEL EIS)

DOE prepared this EIS (DOE 1995d) in compliance with the Court Order dated December 22, 1993, in the case of Public Service Company of Colorado v. Andrus, No. 91-0054-5-HLR (D. Idaho). The preferred alternative presented in the Final EIS, which was issued in April 1995, is Regionalization by Fuel Type. DOE published the Record of Decision, which selected the preferred alternative for L7-2 implementation, on June 1, 1995 (60 FR 28680). Volume 1 of this EIS analyzes at a programmatic level the potential environmental impacts over the next 40 years of alternatives related to the transportation, receipt, processing, and storage of DOE-owned spent nuclear fuel. Volume I supported DOE's programmatic decisions on the sites at which DOE will manage the various types of

DOE-owned spent fuel. The Programmatic Spent Fuel EIS is related to this Interim Management EIS because they both include alternatives for spent fuel currently stored in the SRS reactor disassembly TE basins and the Receiving Basin for Offsite Fuel. Volume I of the programmatic spent fuel EIS is also relevant in the assessment of cumulative impacts that could occur at the SRS during the period evaluated by this EIS. These impacts have been included in the cumulative impact evaluation discussed in Chapter 5.

PROPOSED NUCLEAR WEAPONS NONPROLIFERATION POLICY CONCERNING FOREIGN RESEARCH REACTOR SPENT NUCLEAR FUEL ENVIRONMENTAL IMPACT STATEMENT

On October 21, 1993, DOE announced its intent to prepare this EIS (58 FR 54336), which analyzes TC the acceptance of spent nuclear fuel containing uranium originally produced or enriched in the United States from foreign research reactors. The Draft EIS was distributed in April 1995 (DOE 1995e). A Record of Decision is scheduled for late 1995. The EIS on foreign research reactor spent fuel is relevant to the cumulative impact analysis in this Interim Management EIS because both include alternatives involving the current inventory of highly enriched uranium fuels stored at the Receiving Basin for Offsite Fuel and the reactor disassembly basins at the SRS.

ENVIRONMENTAL ASSESSMENT FOR THE OPERATION OF THE HB-LINE FACILITY AND FRAME WASTE RECOVERY PROCESS FOR PRODUCTION OF PU-238 OXIDE AT THE SAVANNAH RIVER SITE

DOE released this environmental assessment (DOE 1995b) and a Finding of No Significant Impact in TC April 1995. This document addresses the future operation of the HB-Line facility and the Frame Waste Recovery process at the SRS. These facilities process plutonium-238 for energy sources in support of space, scientific, and terrestrial missions. The environmental assessment is related to this EIS because it includes the portion of the current SRS inventory of plutonium-238 that DOE considers usable to meet its programmatic needs. This EIS deals with management alternatives for TE unusable scrap materials that contain plutonium-238. The environmental assessment is also relevant because it evaluates proposed actions that could occur at the SRS during the same period evaluated in this EIS. For this reason, it is relevant in the assessment of potential cumulative impacts (see Chapter 5).

STOCKPILE STEWARDSHIP AND MANAGEMENT PROGRAMMATIC ENVIRONMENTAL IMPACT STATEMENT

On June 9, 1995, the Principal Deputy Assistant Secretary of Energy for Environment, Safety and Health signed a Notice of Intent to prepare this new Programmatic EIS. DOE will use this document to formally introduce the Stockpile Stewardship and Management Program, which includes activities required to maintain a high level of confidence in the safety and reliability of the Nation's nuclear weapons in the absence of underground testing, and to be prepared to resume nuclear testing if so directed by the President. Stockpile Management activities include dismantlement, maintenance, evaluation, and repair or replacement of weapons and their components in the existing stockpile. The scoping period for this EIS ended on August 11, 1995.

UPGRADE OF CANYON EXHAUST SYSTEMS ENVIRONMENTAL IMPACT STATEMENT

On March 20, 1992, DOE published a Notice of Intent in the *Federal Register* to prepare an EIS for the upgrade of the ventilation exhaust system for the canyons at SRS (57 FR 9693). DOE held public scoping meetings in Columbia, South Carolina, Savannah, Georgia, and Aiken, South Carolina on April 21, 24, 28, 1992, respectively. Since that time, DOE has further evaluated the need for upgrades to the canyon ventilation exhaust systems, and has determined that extensive upgrades are not warranted based on safety and cost considerations. Reliability studies and engineering reviews that DOE has conducted since 1992 indicate that only a limited number of components (e.g., exhaust fans) warrant replacement. The estimated cost for the upgrades is approximately $30 million. In light of the substantially reduced scope of required upgrades, DOE has determined that an environmental impact statement is not warranted. L11-11

DRAFT PROGRAMMATIC ENVIRONMENTAL IMPACT STATEMENT FOR TRITIUM SUPPLY AND RECYCLING

DOE is evaluating alternatives for an assured, long-term supply of tritium, a radioactive gas that is a necessary component of every nuclear weapon in the Nation's stockpile. On March 17, 1995, DOE announced the availability of the Tritium Supply and Recycling Draft Programmatic Environmental Impact Statement (60 FR 14433). The EIS evaluates several technologies for the production of tritium, including reactors and an accelerator, and associated support facilities. The SRS is one of the candidate sites evaluated by DOE in the EIS. The public comment period was recently reopened by DOE in response to comments received on the potential use of commercial nuclear reactors to produce tritium (60 FR 44327, August 25, 1995). TC

1.7 Relationship of Decisions

Many of the materials that are Candidates for Stabilization in this EIS are included in the scopes of Programmatic EISs that DOE is preparing (see Section 1.6). These materials include spent fuel and weapons-usable fissile materials such as plutonium-239 or highly enriched uranium. The actions (other than No Action) being considered in this EIS involve either changing the physical form of the nuclear materials or the manner in which they are stored. DOE believes that any actions taken as a result of this EIS would be interim actions (within the context of the National Environmental Policy Act) that are warranted for safety reasons independently of programs for long-term use, management, or disposition (see Sections 1.4 and 1.5.2).

For example, the Programmatic EIS for storage and disposition of weapons-usable fissile materials
TE evaluates alternatives for the disposition of plutonium-239 that is surplus to the Nation's defense
requirements. This Interim Management of Nuclear Materials EIS evaluates alternatives for the safe
storage of various plutonium-239 materials (e.g., solutions in H-Canyon) for a period as long as
TC 10 years. DOE chose this period in response to public comments because it could take that long to
make and begin implementing decisions for the disposition of surplus materials such as
plutonium-239. DOE believes actions taken for safety and health purposes for plutonium-239 are
TE independent of actions that might be required for the disposition of those materials. Further, DOE
believes the physical form of stabilized plutonium-239 at the SRS will neither eliminate disposition
alternatives nor predetermine disposition decisions for the entire DOE inventory of surplus
plutonium-239.

CHAPTER 2. ALTERNATIVES

This chapter identifies the alternatives that DOE has evaluated for each material type and identifies DOE's preferred alternatives. Table 2-1 lists the alternatives. Although most of the alternatives evaluated in this EIS would rely on the use of existing facilities at the SRS, some would require new or modified facilities. This chapter identifies such facilities for each alternative, if applicable. Appendix C contains detailed descriptions of the facilities and their operations.

DOE has identified three broad categories of materials: Stable, Programmatic, and Candidates for Stabilization. In general, DOE proposes to maintain Stable material in its current form, convert Programmatic material to a safer form that could also be used to meet future needs, and stabilize materials that present vulnerabilities from continuing to store the materials in their current forms or locations. A number of steps (i.e., phases) are associated with the implementation of any alternative (other than the No-Action Alternative). The description of each alternative in this chapter includes a chart that shows the sequence and approximate duration of the steps evaluated for implementation; the heavier line on each chart indicates the critical time path for that alternative. These charts reflect the intervals evaluated for environmental impacts; the intervals are, in general, longer than DOE expects it would take to implement the alternatives. DOE used this analysis method because it tends to provide a conservative (i.e., higher impact) estimate of impacts from proposed actions. Section 2.5.3 provides more realistic schedules, which have been integrated to maximize stabilization efficiency. TC

DOE used the following general assumptions in the development of the duration charts:

1. If the physical capability to implement the alternative exists, DOE based the estimated time to process, convert, or repackage the material on a fraction of the historic throughput capacity of the facility and recent operating experience in similar facilities (e.g., plutonium-238 operations in HB-Line). In all cases, DOE tried to provide a conservative but realistic prediction of the time required to ensure adequate estimates of resulting environmental impacts. L11-9

2. If the physical capability to implement the alternative does not exist, DOE used engineering studies to estimate (a) the time it would take to design and construct that capability, and (b) the capacity or throughput capability of the new facility (or modified existing facility). The engineering studies used could be characterized as preliminary conceptual design reports or in some cases, feasibility studies. Again, DOE used conservative estimates of the time

Table 2-1. Alternatives for the management of SRS nuclear materials.

Material	Alternatives							
	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[a]	Vitrification (F-Canyon)	Improving Storage	
Stable material	✔							TC
Plutonium-242	✓	✓	✔		✓	✓		L8-2
Americium and curium	✔[b]		✓		✓	✔[c]		L14-1
Neptunium	✓		✔		✓	✓		
H-Canyon plutonium-239 solutions	✓	✓	✔		✓	✓		
H-Canyon enriched uranium solutions	✓		✓	✔	✓			
Plutonium and uranium stored in vaults	✓	✔[d]	✔[d]		✓	✔[d]	✔[d]	TC
Mark-31 targets	✓	✔	✓		✓	✓	✓	
Mark-16 and -22 fuels	✔		✓	✓	✓		✓	
Other aluminum-clad targets	✔				✓		✓	
Failed TRR fuel and EBR-II slugs[e]	✓	✔	✓		✓	✓	✓	TC

✔ = Preferred alternative.

a. DWPF = Defense Waste Processing Facility.

b. Targets.

c. Solutions.

d. For the plutonium and uranium stored in vaults there are four preferred alternatives. DOE will base its choice of the appropriate alternative for a particular solid on results of the material inspection, as discussed in Section 2.3.3.

e. TRR = Taiwan Research Reactor; EBR-II = Experimental Breaker Reactor-II. TC requirements for the design, construction, and operation of any new capabilities. DOE included any time needed to obtain congressional approval and authorization. For example, a line item project would be required in the Federal budget for the construction of a new facility for the dry storage of irradiated reactor fuel or targets.

3. The implementation of several of the alternatives that would use existing facilities or capabilities would require some level of research and development. This is particularly true for the alternatives that would use the SRS high-level waste processing facilities (including the Defense Waste Processing Facility) to vitrify nuclear materials. These alternatives would require extensive analytical laboratory research to develop new chemical processes or to validate the use of existing processes without extensive equipment or facility modifications. The engineering studies used to evaluate these alternatives were essentially feasibility studies. The technical complexity associated with implementing these alternatives is such that DOE conservatively believes it would take several years to demonstrate their technical feasibility even on a laboratory scale. In addition, the development work could indicate the need for modifications to the high-level waste system. L11-9

4. DOE did not include estimated durations for operational readiness reviews or other preparatory actions for implementation of the alternatives. DOE anticipates that such actions would be a prerequisite, regardless of the alternative selected, and that the level of environmental impacts from such activities would be included in the estimates for the No-Action Alternative because the activities would involve training of personnel, maintenance, equipment checkout, and simulated operations.

The sequence charts for each alternative simply show the sequence of the major activities and conservative duration estimates. They do not represent actual schedules for completing any alternative, and they do not show beginning, end, or intermediate milestone dates. DOE would develop an integrated schedule for the alternatives it selected, in response to Defense Nuclear Facilities Safety Board Recommendation 94-1 (see Section 1.5).

ACTIVITIES INCLUDED IN NO ACTION

Many general activities would be associated with the implementation of any of the alternatives that DOE has evaluated. These activities would occur regardless of the alternative eventually selected. For example, DOE would maintain the facilities in good working condition and would continue to provide utilities (water, electricity, steam, compressed gas, etc.) and services (security, maintenance, TC

TC fire protection, etc.) for each facility. Training activities would ensure that appropriate personnel maintained the skills necessary to operate the facilities and equipment.

DOE would relocate, repackage, or recan materials stored in vaults as necessary to maintain safety. Relocation would include the movement of materials to consolidate storage, allow maintenance, or respond to an imminent safety concern. Repackaging would include removing materials from a damaged storage container and placing them in a new container or placing the damaged container in a larger container. Recanning, which would primarily involve fuel and targets, would entail placing damaged or degraded fuel in metal containers, sealing the containers, and returning them to wet storage. DOE could relocate or consolidate fuel and targets in the basins. Sampling, destructive and nondestructive examination, weighing, visual inspections, and similar activities would determine the physical and chemical condition of the material. Existing solutions would require chemical L11-6 adjustments to maintain their required concentration limits and chemistry controls. DOE would continue ongoing programs for the consolidation of highly enriched uranium at the SRS.

In addition, DOE would continue to correct vulnerabilities associated with various facilities currently used to store nuclear materials. DOE described these actions in a Plan of Action it issued in response to the Spent Fuel Working Group Report (DOE 1994a). One of the main goals of these actions is to reduce the corrosion rate of the fuel and targets stored in the reactor disassembly basins. These actions include the use of portable ion exchangers and the installation of new ion exchangers to improve basin water quality. DOE anticipates that these improvements could prevent the initiation of corrosion at places on the material in the basins where corrosion has not yet occurred. However, DOE does not expect these actions to completely stop corrosion already in progress.

2.1 Stable Material

DOE has determined that the condition of most nuclear material at the SRS is not likely to present a safety concern over the next 10 years and that such material is stable and suitable for continued TE storage. Table A-1 lists each Stable material and identifies the facility in which DOE has stored it.

Because Stable material is suitable for continued storage, no actions are necessary to meet the purpose and need for this EIS. Therefore, the preferred alternative for Stable material is Continuing Storage (No Action). Under this alternative, such material would be managed in its existing form to maintain the health and safety of workers and the public. The selection of this alternative for stable materials TC would not necessarily mean that DOE has decided to store them for 10 years, only that it is safe to do so. Certain stable materials could be useful for existing or future DOE programs. If such situations arise, DOE would perform the appropriate planning, including NEPA review. TC

2.2 Programmatic Material

DOE has determined that it might need some of the nuclear material at the SRS to meet future program missions. The following paragraphs identify the missions for such materials, which Appendix A describes in more detail:

- Plutonium-242, which DOE could use in the nuclear weapons stockpile stewardship program. TC

- Americium-243 and curium-244, which DOE would maintain as a national asset for potential support of research in nuclear medicine, nuclear chemistry, solid-state chemistry, and nuclear physics.

- The higher isotopes of curium (curium-244 through -248) are irreplaceable feedstocks for the production of californium-252, which is used as a neutron source for both military and industrial applications. L9-4

- Neptunium-237, which DOE could use in the production of plutonium-238 to provide a power source for remote terrestrial and space applications.

All the programmatic material already exists at the Savannah River Site. However, none of the material is in a form that DOE could use at present. Almost all of the material is in the form of acidic solutions stored in stainless-steel tanks inside the chemical separations facilities (F- and H-Canyons). The solutions have already been chemically separated into the various isotopic components. For example, one tank contains americium and curium, several tanks contain neptunium, and another contains plutonium-242 (see Section 1.4.2). The continued storage of the programmatic material in a liquid form poses inherent environmental, safety, and health concerns. Radioactive solutions could be released as a result of operational errors, equipment failures, or facility-related accidents. Even if DOE had no potential programmatic use for these materials, it would categorize them as Candidates for Stabilization (see Sections 1.4 and 1.5.2). TC

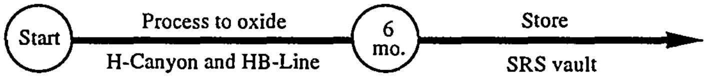
DOE has examined the range of alternatives that could reasonably produce a stable form of material. However, certain alternatives would not only fulfill the objective of producing a stable form of material that DOE could store safely for an extended period of time, but would also result in a form that was more conducive to supporting the potential use of the material. In some cases, an alternative would produce a stable form, but impurities added through the process would render the material useless or make it extremely difficult and costly to recover the programmatic material for potential use at a later date. TC

Future DOE decisions will determine if these materials will actually be used. The Record of Decision following the completion of this EIS will only determine what, if any, stabilization actions DOE will take for these special materials.

2.2.1 PLUTONIUM-242

TC The SRS plutonium-242 that could be used is stored in an aqueous solution in one tank in H-Canyon. DOE has evaluated the following alternatives for the conversion of this plutonium-242 to a form that is safe for continued interim storage.

- **Processing to Oxide.**



TC DOE would convert the existing plutonium-242 solution to an oxide by operating H-Canyon and HB-Line. (Figure 2-1 shows key facilities in H-Area, including the H-Canyon building in the center; the figure also shows the Defense Waste Processing Facility in the adjoining S-Area.) The solutions would be purified as necessary to separate the plutonium-242 from impurities and radioactive decay products in the solution to prepare the material for conversion to a solid in HB-Line. Separated material other than plutonium-242 would be transferred from H-Canyon to the high-level waste tanks via underground pipes. The entire inventory of plutonium-242 solution in H-Canyon would be transferred through pipes to HB-Line, where it would be converted to an oxide. The oxide would be packaged in steel containers and stored in an SRS vault. The material would be monitored and inspected during this storage period but the containers would be opened only to satisfy a concern about safety, material accountability, etc. When the proposed oxide packaging capability in FB-Line or the TE proposed Actinide Packaging and Storage Facility became available (see Appendix C), the existing inventory of material would be evaluated to determine if any action was required to ensure that the material met the DOE standard for storage of plutonium oxides (DOE 1994h).

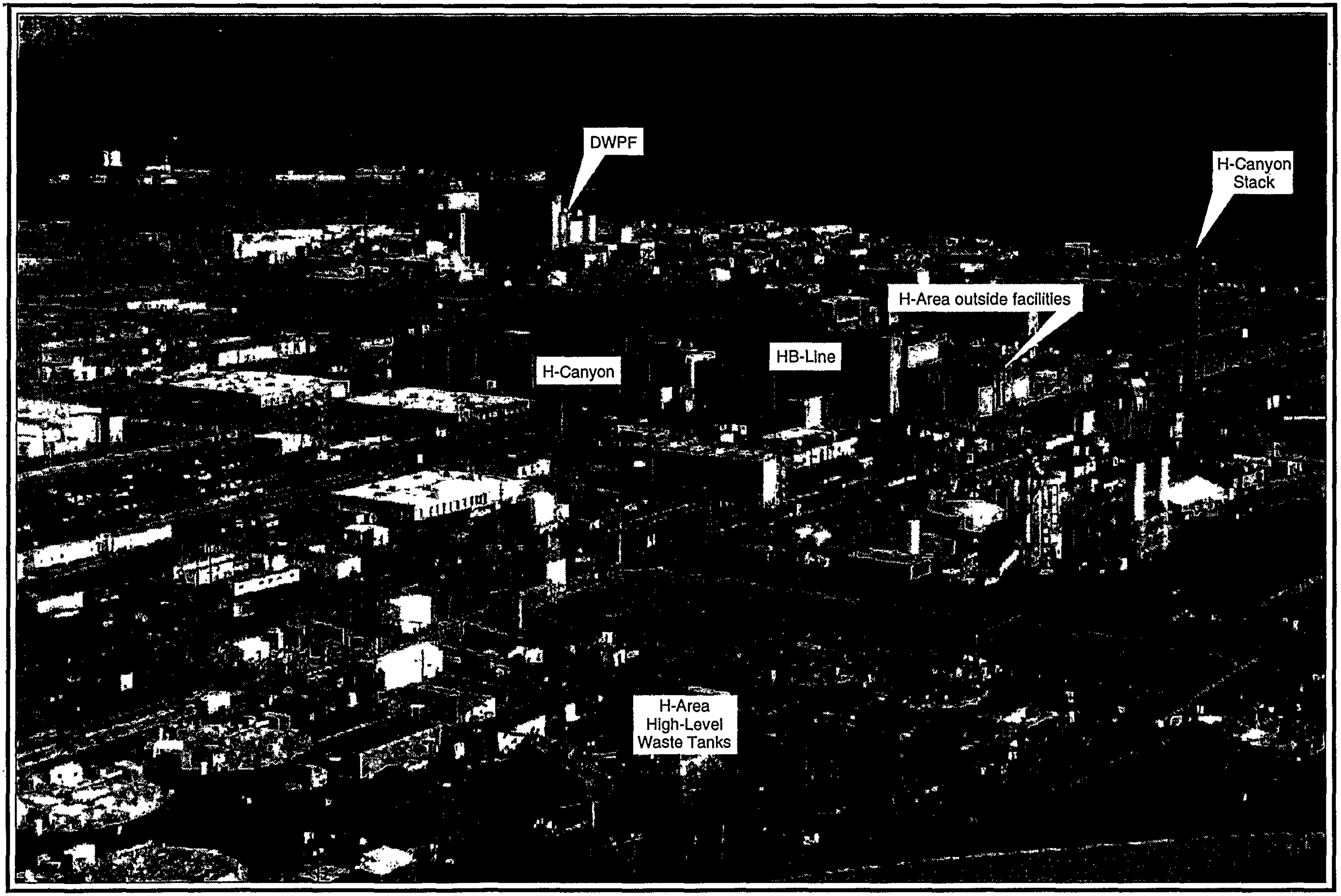
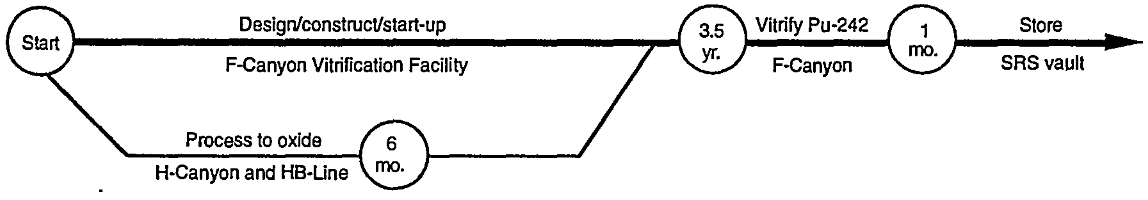


Figure 2-1. H-Area.

TE If actions were required, the material would be transferred to the packaging facility, heated, and repackaged.

- **Vitrification (F-Canyon).**

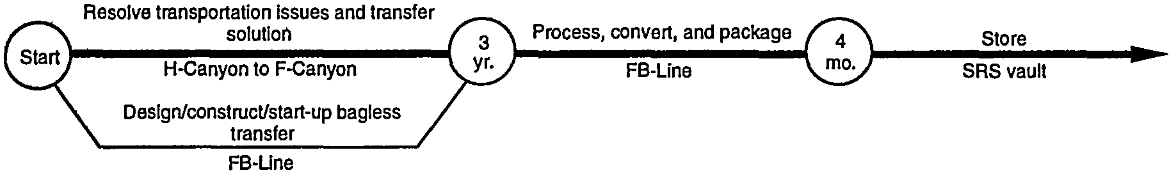


Under this alternative, DOE would be required to add a vitrification facility (see Appendix C). After evaluating the time required to construct this facility, DOE determined the optimum case would be to modify a portion of the F-Canyon. This existing facility would provide the support structure, utilities, and services necessary for vitrification, and only new equipment installation would be required. The time estimated to install the vitrification equipment and make it ready for operation would be about 3-1/2 years. This would be minimized by using knowledge gained from Defense Waste Processing Facility (DWPF) vitrification development and testing, by accomplishing as much design and development work as possible in parallel with other National Environmental Policy Act evaluations, and by using standardized equipment as much as possible. Most of the waste generated from the modification operations would be low-level radioactive waste, which DOE would dispose of in existing SRS disposal facilities. TC

After the facility became operational, DOE would transfer oxide from H-Canyon (produced as described above for the Processing to Oxide Alternative) to F-Area, vitrify it in the F-Canyon Vitrification Facility, and store the canisters in F-Canyon or a shielded vault. As a variation, DOE could transfer the plutonium-242 solutions by truck or rail to F-Area using an appropriate shipping container. At present, however, DOE does not have the capability to make such transfers. DOE would store the plutonium-242 solution in H-Canyon until it could develop or acquire a container suitable for transporting the material to F-Canyon. At present, there is no domestic container licensed for the transport of liquid plutonium by DOE, the Nuclear Regulatory Commission (NRC), or the Department of Transportation (DOT). In fact, current U.S. regulations prohibit the transport of such quantities of plutonium in liquid form on public thoroughfares. However, because the movement of the material would be inside the SRS boundaries, DOE could approve the transport if the workers and the public were adequately protected. This would require the use of an extremely robust container, similar to those used for the shipment of spent fuel. DOE has evaluated the availability of containers internationally and has identified at least one as potentially usable. The container is licensed by the International Atomic Energy Agency (IAEA), but not for the quantities or isotopic compositions of the plutonium-242 solution or the other solutions stored in the canyons. DOE would have to perform a technical evaluation of the design to ensure its adequacy. DOE could either procure the container or design, fabricate, and license its own container. In either case, DOE would have to develop a method to accomplish the intra-area transport of the solution from H-Canyon to F-Canyon. TC

After transporting the solution to F-Area, DOE could move it into F-Canyon by using a transfer line in the F-Area Outside Facilities east of the canyon or by bringing the shipping container into the canyon and transferring the solution to process vessels. Other transfer methods could be utilized, such as introducing the material through FB-Line. When the material was in the facility, it would be purified, if required, to ensure the purity of the plutonium-242. TC The material would be chemically adjusted as required to meet the specifications for introducing the plutonium to the vitrification process, and directed through intrafacility piping to the vitrification facility where the plutonium would be combined with TE molten glass, poured into steel containers, cooled, and placed in storage in the canyon or a shielded vault. High-level waste generated during these operations would be transferred to the F-Area high-level waste tanks.

- **Processing to Metal.**

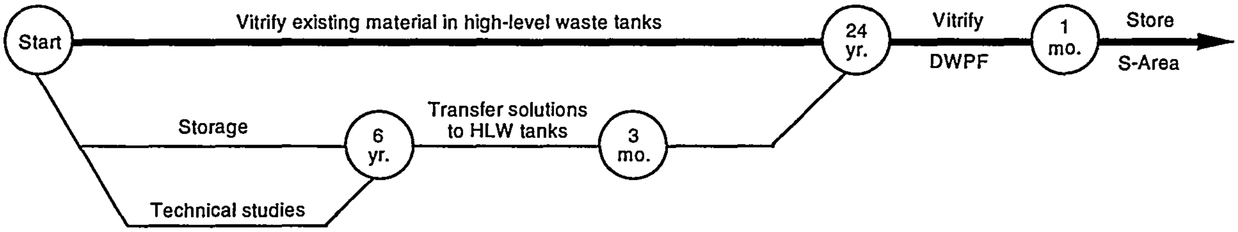


TC

DOE would transport the solution from H-Canyon to F-Canyon, in a manner similar to that described for the Vitrification (F-Canyon) Alternative. In F-Canyon, the solutions would be purified using portions of the PUREX process and the material would be transferred to the FB-Line. The purified solution containing the plutonium-242 would be converted to a metal, packaged, and placed in a storage vault. Chemically separated material other than the plutonium-242 (e.g., impurities) would be transferred to the high-level waste tanks in F-Area via underground pipes. As with the Processing to Oxide Alternative, the metal would be

TC monitored and inspected while in storage. In parallel with this effort, DOE would begin modifications to FB-Line to provide the capability to package plutonium in accordance with the Departmental standard (DOE 1994h). A glovebox would be added to enable repackaging of the material in a nonreactive atmosphere without the use of plastic wrapping. After the modifications, DOE would repackage the metal and return it to storage in an F-Area vault.

- **Processing and Storage for Vitrification in the Defense Waste Processing Facility.**



DOE would continue to store the plutonium-242 solutions until the completion of technical feasibility studies. These studies would be necessary to determine the potential magnitude of the plutonium-242 contribution to saltstone radioactivity and assess whether the resulting saltstone radioactivity would exceed permitted limits. When the studies were complete, DOE L11-5 would adjust the solution chemically as necessary for discharge to the waste tanks. DOE would vitrify the plutonium-242 at the Defense Waste Processing Facility after it had processed the existing inventory in the waste tanks, which will take approximately 24 years (WSRC 1994a). The physical configuration (i.e., serial relationship) of the high-level waste tanks would not support vitrification of the solutions until the DWPF had processed the existing inventory of high-level waste. This alternative would make the material unavailable to meet the programmatic need because, once the material was discarded to the high-level waste tanks, it would be mixed with all other waste and diluted so much that it would be unrecoverable.

Throughout this EIS, DOE has evaluated the potential processing and storage of nuclear materials for eventual vitrification using the DWPF. DOE included this alternative because of acknowledged stakeholder interest. Although technically feasible, the complexity of the alternative renders it unlikely for successful implementation within the next 10 years.

- **Continuing Storage (No Action).** DOE would continue to store the plutonium-242 solution in the H-Canyon tank.

2.2.2 AMERICIUM AND CURIUM

About 14,000 liters (3,800 gallons) of americium and curium solution are stored in a single tank in F-Canyon (Figure 2-2 shows F-Area with the F-Canyon building in the center). Sixty-five Mark-18 assemblies and 174 slugs are stored in the Receiving Basin for Offsite Fuel (RBOF) and the reactor disassembly basin (see Table 1-1); these materials contain additional valuable amounts of americium and curium. The alternatives and impact information discussed in this EIS for these materials include the solution, assemblies, and slugs.

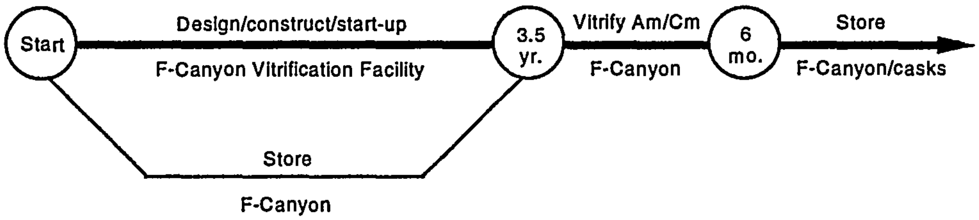
L9-1
L9-5
L10-1
L13-1
L14-9

Americium and curium are feed materials in the DOE National Heavy Element Program that produces a large number of heavier transuranium elements, including californium-252. Californium-252 has a wide variety of medical, commercial, and defense-related uses, which include cancer treatment and treatment research, neutron radiography for nondestructive testing of metal parts in aircraft, the online assay of coal to comply with DOE emission regulations, and online quality control in cement manufacture.

To be suitable for potential programmatic use, the material should be in a solid form that could be transported to and used by the Oak Ridge National Laboratory (the DOE user). DOE would have to convert the americium and curium solution in F-Canyon to a solid. Similarly, DOE could dissolve and process the solid targets and slugs to a concentrated solid form for safe interim, storage and potential use. L14-9

DOE has identified the following alternatives for evaluation in considering conversion of the americium and curium material to a form suitable for continued interim storage.

- **Vitrification (F-Canyon).**



DOE would continue to store the material in F-Canyon and the basins while undertaking studies, design work, and modification of a portion of the canyon to add a vitrification capability. DOE would create the vitrification facility by modifying an area inside the hot L14-9

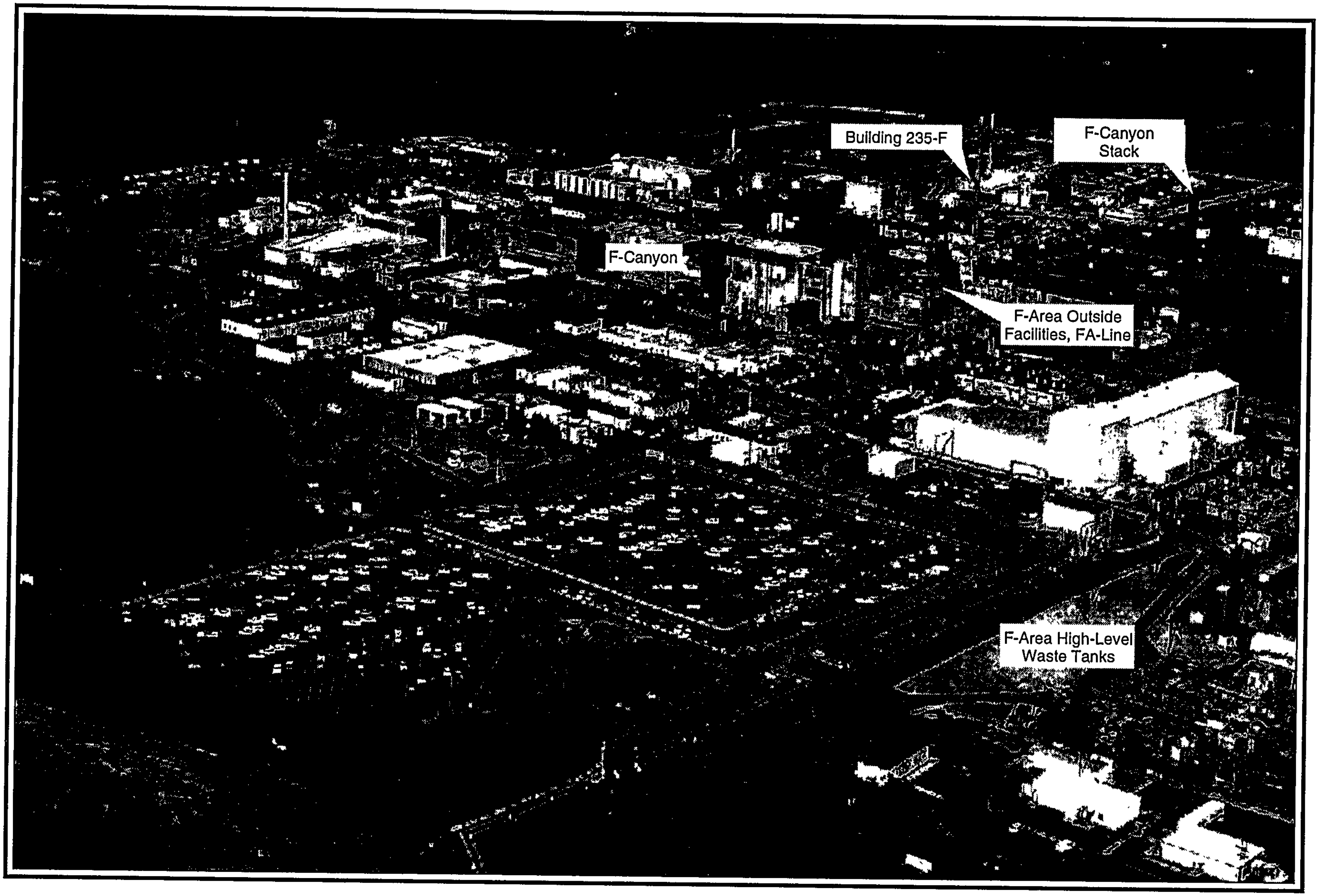
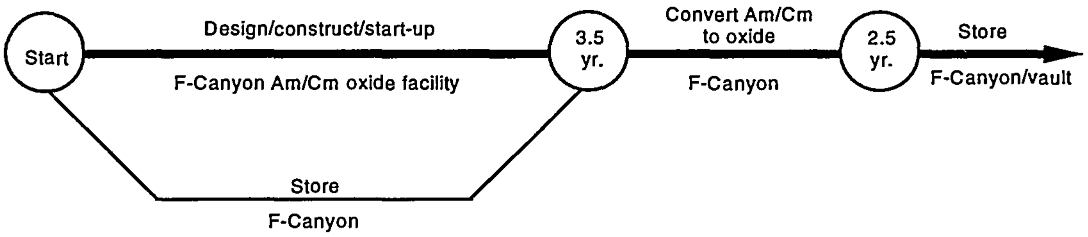


Figure 2-2. F-Area.

canyon (explained in Appendix C). This modified area - the F-Canyon Vitrification Facility - would take about 3-1/2 years to complete. Most of the waste generated from the modification operations would be low-level radioactive waste, which DOE would dispose of in existing SRS disposal facilities.

After the facility became operational, DOE could dissolve the targets and slugs. DOE would process the americium and curium solution to remove impurities and radioactive decay products and chemically adjust the material as necessary to meet vitrification process feed requirements. Then the material would be transferred to the vitrification facility. DOE would vitrify the material, pour it into stainless-steel canisters, seal the canisters, and place them in storage at the SRS. DOE expects it would take about 6 months to vitrify the americium and curium solutions, producing about 40 canisters. The radiation level would be very high, about 90 rem per hour at 1 meter (3.2 feet) from a canister. High-level waste generated from chemical processing operations would be transferred to the F-Area high-level waste tanks. L14-9

- **Processing to Oxide.**

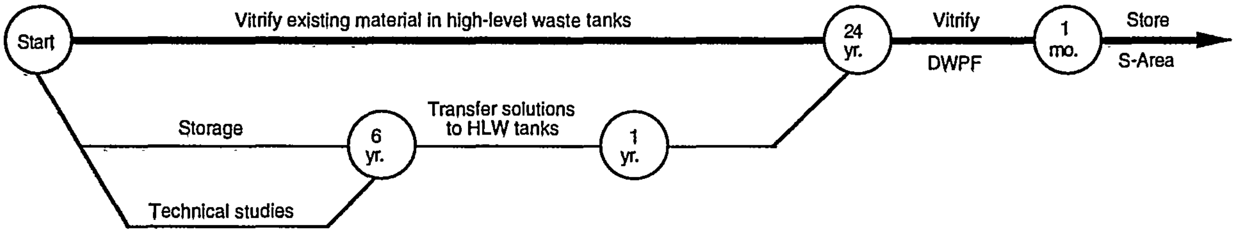


DOE would continue to store the material in F-Canyon and the basins while undertaking studies, design work, and modification of a portion of the canyon to add the capability to process americium and curium to oxide. These modifications would take about 3-1/2 years to complete. A problem associated with oxide production is that the operation of the process would be limited to batches of 500 grams (17.6 ounces). Larger quantities would cause self-heating of the material to an extent that would impede the oxide conversion process. At this rate, it would take about 2-1/2 years to convert all the americium and curium to oxide, even if DOE operated the conversion facility 24 hours a day, 7 days a week. This operation would yield about 250 cans of americium and curium oxide. Another problem is that the americium and curium oxide would emit very high levels of radiation. Each can of oxide could produce radiation levels as high as 30 rem per hour at 1 meter (3.2 feet). As a result, all loading and packaging operations (which are normally performed by hand) would have to be accomplished remotely. Designs for this remote operation would be complicated, particularly L14-9 TE

TE for oxide powder handling, and would be the factor of greatest uncertainty associated with the implementation of this alternative.

L14-9 After the facility became operational, DOE could dissolve the targets and slugs and purify the americium and curium solution to remove impurities and radioactive decay products and chemically adjust the material as necessary to meet the oxide conversion process feed requirements. Then the solution would be transferred through pipes inside the canyon to the oxidation facility. The material would be converted to an oxide, sealed in containers, and placed in appropriate storage canisters. The material would be stored in F-Canyon or transferred to a heavily shielded vault for storage. High-level waste generated during processing would be sent to the F-Area high-level waste tanks via underground pipes.

- **Processing and Storage for Vitrification in the Defense Waste Processing Facility.**



DOE would continue to store the americium and curium solutions until technical feasibility studies were complete. These studies would be necessary to determine the potential magnitude of the americium and curium contribution to saltstone radioactivity and assess whether the resulting saltstone radioactivity would exceed permitted limits. When the studies were complete, DOE could dissolve the targets and slugs and adjust the resulting solutions chemically as necessary for discharge to the waste tanks, and would eventually vitrify the material at the Defense Waste Processing Facility. The implementation of this alternative would make the americium and curium unavailable to meet the programmatic need because, once the material was discarded to the high-level waste tanks, it would be mixed with all other waste and diluted so much that it would be unrecoverable. In addition, the increased radiation levels likely to be generated by introducing this material to the high-level waste tanks could be reduced to "historic or average" levels only by diluting the waste volume with an additional 3,785,000 liters (1,000,000 gallons) of liquid waste. L11-5

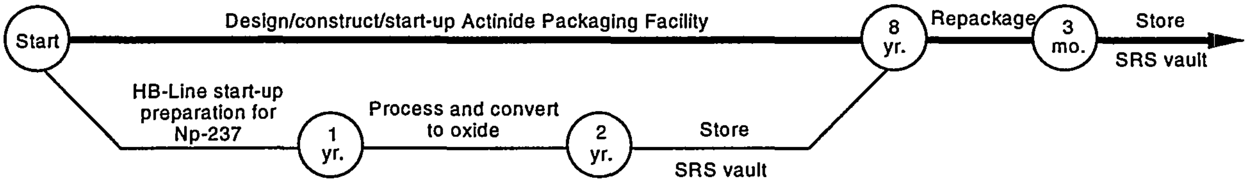
- **Continuing Storage (No Action).** DOE would continue to store the americium and curium solution in F-Canyon and the target assemblies and slugs in the Receiving Basin for Offsite Fuel. DOE would relocate the slugs in the P-Reactor disassembly basin to RBOF. L11-5

2.2.3 NEPTUNIUM-237

Approximately 6,100 liters (1,600 gallons) of neptunium-237 solution are currently stored in H-Canyon storage tanks. In addition, nine neptunium targets are stored in M-Area. The alternatives and impact information discussed in this EIS for this material include the solution and targets.

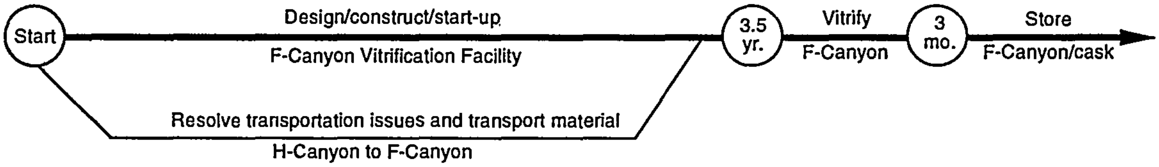
Neptunium-237 is used in the production of plutonium-238, the principal use of which is in thermal power generators in applications where solar power or chemical batteries are not practical, such as exploratory spacecraft. DOE has identified the following alternatives for evaluation in considering conversion of the neptunium-237 in targets and solution to a form for safe interim storage and potential use. TC

- **Processing to Oxide.**



DOE would begin by transferring the nine neptunium targets from M-Area to H-Canyon and dissolving them. This material would be processed through the canyon and added to the existing neptunium solution. Although the targets in M-Area are in oxide form now, they were fabricated originally to be irradiated in the SRS reactors and cannot be used in other reactors in their current form. The SRS reactors are no longer operating. The oxide form TC would be such that future users (e.g., Oak Ridge National Laboratory) could fabricate the type of target required for their plutonium-238 production processes. DOE would purify the neptunium solution as required to remove radioactive decay products and other chemicals that could interfere with the oxide conversion process. The resulting neptunium solution would be transferred to the HB-Line through intrafacility pipes and converted to neptunium oxide. The radioactive decay products and other material would be transferred through underground pipes to the high-level waste tanks. The oxide would be put in shielded containers and placed in storage in an F-Area vault. When the proposed Actinide Packaging and Storage Facility TE became available or the proposed FB-Line modifications for oxide packaging were completed (see Appendix C), any material that had not been used for programmatic purposes would be heated and repackaged if required to ensure long-term stability.

- **Vitrification (F-Canyon).**

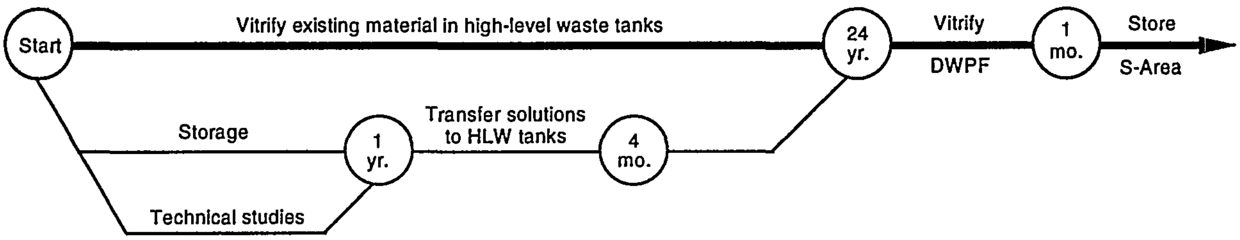


TC

DOE would continue to store the neptunium solution in H-Canyon and the targets in M-Area. During this time, DOE would complete the necessary technical evaluation to determine the feasibility of obtaining a container that would enable the shipment of neptunium solutions across the SRS. In addition, DOE would undertake the studies, design work, and required equipment changes to provide the capability to vitrify neptunium-237 in F-Canyon (see Appendix C). Then DOE would transfer the neptunium-237 targets and solution to F-Canyon or FB-Line, using an appropriate shipping container (truck or rail). At present, DOE does not have the capability to make such transfers. The issues of container certification and availability must be resolved. In F-Area, the material could be moved into F-Canyon by using a transfer line in the F-Area Outside Facilities east of the canyon or by bringing the shipping container into the canyon and transferring the solution or targets to process vessels. Other transfer methods could be utilized, such as introducing the material through FB-Line. When the material was in the facility, it would be purified as required to remove radioactive decay products and other impurities. The material would be chemically adjusted as required to meet the specifications for introducing the neptunium to the vitrification process. The material would be directed through intrafacility piping to the vitrification facility where the neptunium would be combined with molten glass, poured into steel containers, cooled, and placed in storage in the canyon or a shielded vault. High-level waste generated during these operations would be transferred to the F-Area high-level waste tanks.

TC

- **Processing and Storage for Vitrification in the Defense Waste Processing Facility.**



DOE would continue to store the neptunium solutions until technical feasibility studies were complete. These studies would be necessary to determine the potential magnitude of the neptunium contribution to saltstone radioactivity and assess whether the resulting saltstone radioactivity would exceed permitted limits. When the studies were complete, DOE would adjust the resulting solution chemically as necessary for discharge to the waste tanks and would eventually vitrify the material at the Defense Waste Processing Facility. The implementation of this alternative would make the neptunium unavailable to meet the programmatic need because, once the material was discarded to the high-level waste tanks, it would be mixed with all other waste and diluted so much that it would be unrecoverable. L11-5

- **Continuing Storage (No Action).** DOE would continue to store the neptunium solution in H-Canyon and the targets in M-Area or another suitable storage facility on the Site.

2.3 Candidate Materials for Stabilization

DOE would stabilize a material if its physical form or storage configuration was a safety concern, or if it could become a safety concern within the next 10 years, or to correct vulnerabilities identified with continued storage of the material. TC DOE evaluated a range of alternative stabilization methods for each category of nuclear material, and used the following criteria to select the alternative stabilization methods for evaluation:

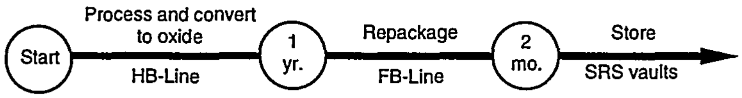
- The product of the proposed action should be stable over a reasonable period of time to prevent the need to restabilize the material.

- The stabilization method should involve a technology that would enable the initiation of stabilization actions as quickly as practical and within the period covered by this EIS.

2.3.1 H-CANYON PLUTONIUM-239 SOLUTIONS

Approximately 34,000 liters (9,000 gallons) of plutonium nitrate solutions are stored in stainless-steel tanks in the H-Canyon facility. DOE has identified the following alternatives for management of these solutions.

- **Processing to Oxide.**



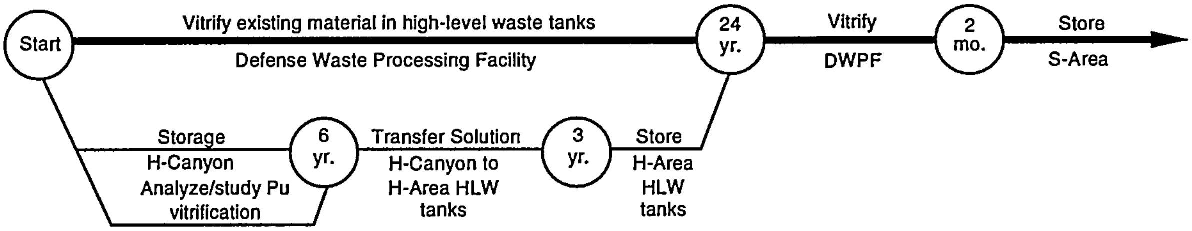
TE DOE would process the plutonium-239 solutions by operating H-Canyon as necessary to
remove radioactive decay products and other impurities that would interfere with subsequent
stabilization steps. No actions would occur to achieve a specific purity for the plutonium in
the solution other than those necessary to operate the process. DOE would transfer the
separated impurities to the H-Area high-level waste tanks, and would transfer the plutonium
TE solutions to the HB-Line for conversion to an oxide. DOE would place the oxide in storage
containers, load the containers in shipping containers, and transport the material to F-Area for
storage. In parallel with this effort, DOE would modify a portion of the existing FB-Line to
provide the capability to package plutonium oxide in a manner that met the storage criteria the
TE Department has established for plutonium oxides (DOE 1994h). A glovebox would be added
to FB-Line to enable the oxide to be heated and packaged in a nonreactive atmosphere without
the use of plastic wrapping material. After the completion of the FB-Line modifications, DOE
would transfer the plutonium oxide to that facility, heat it to meet long-term storage criteria,
package it, and transfer it to a storage vault in F-Area.

If DOE determined that it could not modify the FB-Line to provide the proper packaging
capability or the capability for future inspection and packaging maintenance, it would begin
work on the proposed Actinide Packaging and Storage Facility (see Appendix C); this would
occur in parallel with plutonium conversion activities, but the facility would take about 8 years
TC to complete and begin operations. DOE would transfer any plutonium oxide that was not
packaged to meet the plutonium storage criteria to this facility and repackage it.

DOE has identified actions that could accelerate the construction and startup of the Actinide Packaging and Storage Facility. The normal project management process for a facility that will be a major systems acquisition (i.e., more than $100 million) has about seven steps including four "key decision" points, which range from the initial steps of identifying facility functional design and performance criteria to the final step of approving the facility for operation. DOE could consolidate several of the initial project steps to integrate and accelerate design and construction work through the use of design and construction proposal packages. These consolidation actions could reduce design and construction time such that the facility could be ready for operation in about 6 years. TC

The Actinide Packaging and Storage Facility or the modifications to FB-Line would provide the capability to package plutonium oxide (or plutonium metal) to meet recent Departmental recommendations for the safe storage of plutonium metal and oxides (DOE 1994h). For plutonium oxides, the recommended packaging criterion is that the material be heated to achieve a condition where less than 0.5 percent of the weight of the material is lost by subsequent heating (over a specified time period) and that, following the heating step, the material is cooled and packaged for storage in a nonreactive atmosphere so the benefits of the heating step are retained. The purpose of these actions is to minimize the amount of gas generated within the container used to store the material because the gas has the potential to pressurize, and occasionally cause failure of, a storage container. Gas, normally oxygen and hydrogen, could be generated from the decomposition of water molecules by the radiation given off by the plutonium. The new heating and packaging steps would substantially reduce the amount of moisture in the plutonium oxide, thus reducing potential gas generation. For metal, the criterion is to package the material in a nonreactive atmosphere with no contaminants such as plastic wrapping. The existing B-Line facilities at the SRS (where packaging traditionally occurred) do not have the equipment required to accomplish these new steps.

- **Processing and Storage for Vitrification in the Defense Waste Processing Facility.** TE



DOE would continue to store the H-Canyon plutonium solution until it was ready to discharge it to the H-Area high-level waste tanks. The material would eventually be vitrified at the Defense Waste Processing Facility.

The DWPF was designed to process 132.5 million liters (35 million gallons) of high-level waste (currently stored in F- and H-Area waste tanks) into a glass material encased in stainless-steel cylinders that would be suitable for disposal in a geologic repository. The first step for vitrifying the H-Canyon plutonium solutions would be to transfer the solutions to the high-level waste tanks, which will feed the DWPF. Before transfer, DOE would adjust the solutions to ensure the nuclear criticality safety of the material in the tanks. DOE has identified several concepts for such adjustments: diluting the solutions with water and chemicals to achieve very low fissile material concentrations, diluting the solutions with depleted uranium, or adding iron and manganese or other neutron poisons such as gadolinium (DOE 1994e). After transfer to the waste tanks, the material would be stored and eventually transferred to the DWPF for vitrification. TE

For DOE to use the existing high-level waste treatment processes at the SRS as a stabilization method for nuclear materials, it would have to overcome several technical problems. These waste facilities were designed specifically to accommodate and treat the high-level liquid waste produced by the chemical separations facilities (i.e., F- and H-Canyons). The canyon processes dissolved irradiated materials and chemically separated and recovered selected nuclear materials, primarily plutonium and uranium (see Appendix C for detailed facility descriptions). Secondary canyon processes included the capability for the chemical separation of other desirable isotopes such as neptunium, americium, and curium. The canyon processes were designed for maximum efficiency in the recovery of the desired materials or special isotopes before the other separated materials (aluminum, fission products, etc.) were transferred in the form of radioactive liquids to the high-level waste facilities (see Appendix C). L11-10

The canyon processes were designed to ensure the transfer of only trace quantities of the nuclear materials or special isotopes with the liquid waste stream. This was especially true for fissile isotopes such as plutonium-239 and uranium-235. Because the canyons were so efficient in recovering these fissile isotopes, the high-level waste systems and facilities did not have to accommodate them. This means that the equipment and systems in the high-level waste systems were not designed to be geometrically safe and do not physically preclude the potential accumulation of critical masses of fissile material. Rather, these systems rely on the canyon facilities to ensure that fissile materials are present in only trace quantities in the liquid wastes received. In this manner, the high-level waste, even when stored in million-liter tanks, could not pose a criticality hazard.

For DOE to implement the Processing and Storage for Vitrification at the Defense Waste Processing Facility Alternative successfully for the nuclear materials in this EIS, it would have to review the design of all the processes and facilities used to handle or treat high-level liquid radioactive waste. DOE would have to develop or design a process that could render the fissile materials incapable of producing a nuclear criticality, regardless of the location or amount accumulated in the various equipment or tanks. As indicated above, technical experts at the SRS have proposed several ideas on how to accomplish this. It could be accomplished by the addition of a chemical or other material to serve as a nuclear "poison," precluding or minimizing the potential for a criticality. However, the nuclear poison would have to be designed to accompany the fissile material throughout the process, or different poisons would have to be used at different steps (evaporation, concentration, precipitation, and ultimately vitrification).

Not only are there technical obstacles involving fissile materials that DOE would have to overcome, there are similar problems with other nuclear materials or special isotopes such as americium and curium. Although some of these materials or isotopes might have been transferred to the high-level waste system during the past 40 years, they are present only in trace quantities. The radioactive solutions and nuclear materials discussed in this EIS contain highly concentrated forms of isotopes such as neptunium-237, americium-243, and curium-244. Due to the high concentrations, DOE cannot simply transfer these solutions (or dissolve the materials and transfer the resulting solutions) to the high-level waste system. As with the fissile materials, DOE must evaluate the impacts some isotopes could have on the high-level waste system. Dilution of the solutions could be the only method by which the high-level waste system could accommodate some materials. Dilution could be achieved by mixing solutions with the existing volume of liquid waste or by diluting the solutions into batches in the canyon tanks before transferring the material to the high-level waste system. Either method could require several years. DOE must evaluate the implications of dilution activities on long-term waste volumes and the operation of the associated facilities (e.g., the number of glass canisters produced in the Defense Waste Processing Facility). DOE has included an estimate of such impacts, based on the current level of knowledge and engineering judgment. L11-10

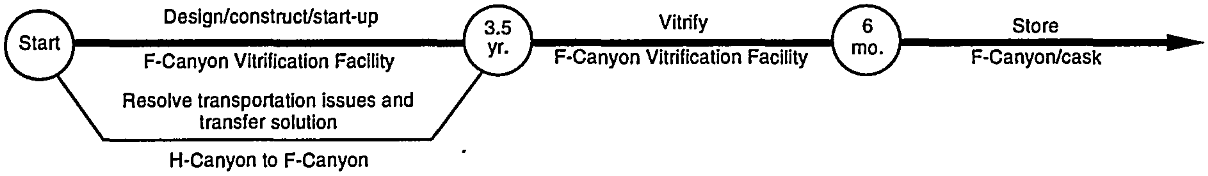
DOE has examined this alternative in some detail and 6 years is the conservative estimate for resolving all of the technical issues. The technical complexity of the alternative is such that the number of unknowns is large. For example, DOE could find that physical modifications would be required to portions of the high-level waste system and the Defense Waste Processing Facility. Given the technical difficulty encountered in the development of the DWPF process, DOE considers the technical uncertainty associated with this alternative to be extremely high. DOE estimates that it would need 3 years or more to make chemical adjustments and transfer these solutions to the waste tanks. The 3-year estimate includes the potential need to dilute the solutions (to reduce the concentration of fissile or special isotopes) and make gradual transfers in batches. The 3-year period also includes time to observe actual versus theoretical or estimated effects based on laboratory development work and models.

DOE considered the potential for transporting solutions of nuclear materials directly from the two canyons and introducing them directly into the DWPF process for vitrification. This would alleviate the need to transfer the solutions to the waste tanks and the intermediate handling and treatment facilities. This variation on the DWPF vitrification alternative would require DOE to either design or procure a method for transporting the radioactive solutions. No shipping containers are available domestically for this use. DOE evaluated a number of shipping containers that are available internationally and that could be used, and identified at least one container. However, no containers are currently certified by the International Atomic Energy Agency (IAEA) for the transportation of solutions with the isotopic composition and in the quantities of those stored at the SRS (or of those that would be produced by the dissolution of additional materials). DOE would have to procure the rights to such a container and further evaluate the feasibility of using or modifying it to support the transportation required. L11-10

Besides developing a method to transport the solutions, DOE would have to design and make modifications to the canyon facilities to enable the loading of the container(s). Similarly, DOE would have to design and make modifications to the Defense Waste Processing Facility to enable the introduction of material directly to the process. This could require either the pretreatment of the solutions before their transport or additional pretreatment equipment in the DWPF. Laboratory research would be required to determine the effects the materials would have on the existing DWPF process and to develop required modifications to the process or the equipment. Equipment changes or modifications to the DWPF process could be required because of the amounts of fissile material or high-activity isotopes that would be in the glass matrix. The DWPF canisters were designed to accommodate specific loadings of radioactive and fissile material. All of the solutions that would require stabilization or that would result as an intermediate step in the vitrification alternative could exceed the design limits of the DWPF glass and canisters. The design and certification of the canisters could require revision.

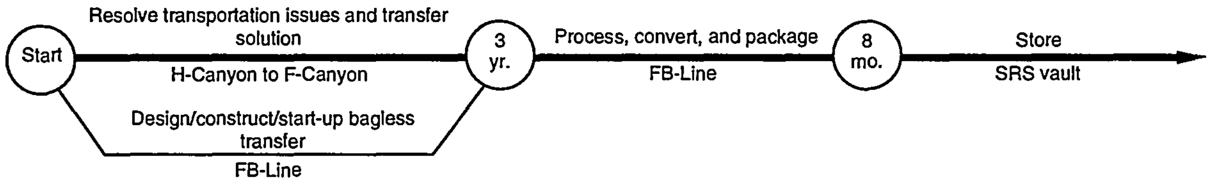
Because direct transportation of the solutions to the DWPF would involve so many issues that DOE would have to resolve, the technical uncertainty with this variation would be similarly high. DOE estimates that this variation would require similar amounts of time for development work, design, modification of facilities, and implementation. Besides the other technical issues, DOE does not believe it would be practical to modify the existing DWPF process to accommodate the direct introduction of the fissile materials or special isotopes until the DWPF process has been demonstrated technically and the existing volumes of high-level waste have been vitrified. L11-10

- **Vitrification (F-Canyon).**



DOE would complete the necessary technical evaluation to determine if it would be feasible to obtain a container suitable to enable the shipment of plutonium solutions across the SRS. At present, DOE does not have the capability to make such transfers. The issues of container certification and availability must be resolved. In addition, DOE would undertake the studies, design work, and equipment changes required to provide the capability to vitrify plutonium in F-Canyon (see Appendix C). Then DOE would transfer the H-Canyon plutonium solution to F-Canyon or FB-Line, using an appropriate shipping container (truck or rail). In F-Area, the material could be moved to F-Canyon by using a transfer line in the F-Area Outside Facilities east of the canyon or by bringing the shipping container into the canyon and transferring the solution to process vessels. Other transfer methods could be used, such as introducing the material through FB-Line. When the material was in the facility, it would be processed by chemical separation and chemically adjusted as required to meet the specifications for introducing the plutonium to the vitrification process. The material would be directed through intrafacility piping to the vitrification facility where the plutonium would be combined with molten glass, poured into stainless-steel canisters, cooled, and placed in storage in the canyon or a shielded vault. High-level waste generated during these operations would be transferred to the F-Area high-level waste tanks.

- **Processing to Metal.**



DOE would complete the necessary technical evaluation to determine the feasibility of obtaining a container that would enable the shipment of plutonium solutions across the SRS. At present, DOE does not have the capability to make such transfers. The issues of container certification and availability must be resolved. Then DOE would transfer the H-Canyon plutonium solution to F-Canyon or FB-Line, using an appropriate shipping container (truck or rail). In F-Area, the material could be moved into F-Canyon by using a transfer line in the F-Area Outside Facilities east of the canyon or by bringing the shipping container into the canyon and transferring the solution to process vessels. Other transfer methods could be used, such as introducing the material through FB-Line. When the material was in the facility it would be processed via chemical separation as required to meet the specifications for introducing the plutonium to FB-Line. No actions would occur to achieve a specific purity for this material other than those necessary to operate the process. The solution would be transferred through the FB-Line process equipment and converted to metal buttons. The buttons would be packaged and stored in an F-Area vault. Any high-level waste generated during this process would be transferred to the F-Area high-level waste tanks. In parallel with this effort, DOE would begin modifications to FB-Line to provide the capability to package plutonium metal in accordance with the Departmental plutonium storage standard (DOE TE 1994h). A glovebox would be added to the FB-Line facility to enable the material to be packaged in a nonreactive atmosphere without the use of plastic wrapping material. After the modifications, DOE would transfer the plutonium metal there and package it to meet DOE storage requirements for plutonium metal (i.e., the metal would be cleaned and repackaged in a nonreactive atmosphere and sealed in a container). The packaged material would be placed in an F-Area vault.

If DOE determined that it could not modify the FB-Line to provide the proper packaging capability or the capability for future inspection and packaging maintenance, DOE would TE begin work on the proposed Actinide Packaging and Storage Facility; this would occur in parallel with plutonium conversion activities, but the facility would take about 8 years to complete and begin operations.

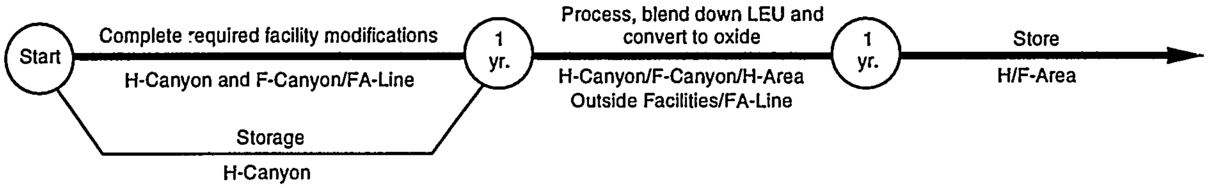
- **Continuing Storage (No Action).** DOE would continue to store the plutonium-239 solution in H-Canyon. TE

2.3.2 H-CANYON URANIUM SOLUTIONS

TE

There are approximately 228,000 liters (60,000 gallons) of enriched uranium nitrate solutions in stainless-steel tanks inside and outside the H-Canyon facility. DOE has identified the following alternatives for management of these solutions.

- **Blending Down to Low Enriched Uranium.**

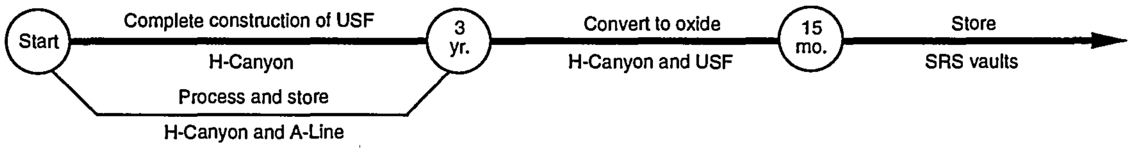


Before stabilizing the enriched uranium, DOE would purify the solutions in H-Canyon to separate the enriched uranium from the other material in the solution (e.g., radioactive decay products normally present in irradiated fuel). DOE would transfer the radioactive decay products and other material to the H-Area high-level waste tanks. DOE would stabilize the highly enriched uranium solution (composed of approximately 60 percent uranium-235) by converting the material to uranium oxide. TC

The FA-Line is the only SRS facility designed to produce uranium oxide, but it was not designed to produce oxide from solutions of highly enriched uranium. To use the FA-Line, DOE would dilute the uranium-235 solution with existing depleted uranium. DOE would accomplish this by dissolving the depleted uranium oxide in FA-Line. DOE would transport the depleted uranium solution to H-Canyon by truck and blend it with the enriched uranium solution to achieve a diluted solution of uranium-235. DOE would transport the mixture back to FA-Line by truck and convert it to low enriched uranium oxide. The final product would be loaded into 208-liter (55-gallon) drums for storage. DOE would make minor modifications in F- and H-Areas to enable truck-trailer loading and unloading and to install a spare oxide dissolver at FA-Line. In addition, DOE would construct a storage facility with an area of approximately 186 square meters (2,000 square feet) on previously disturbed land in the industrialized F-Area to handle the drums of uranium oxide. TC

A variation of this alternative would be to transport the uranium solution from H-Area to F-Area by rail or truck using an appropriate shipping container. FA-Line would be used to dissolve depleted uranium oxide and blend it with the uranium solution from H-Area to achieve a low enriched uranium solution. Blending operations could occur in F-Canyon process vessels or in F-Area Outside Facility tanks. The facility modifications and the storage facility described above would be required.

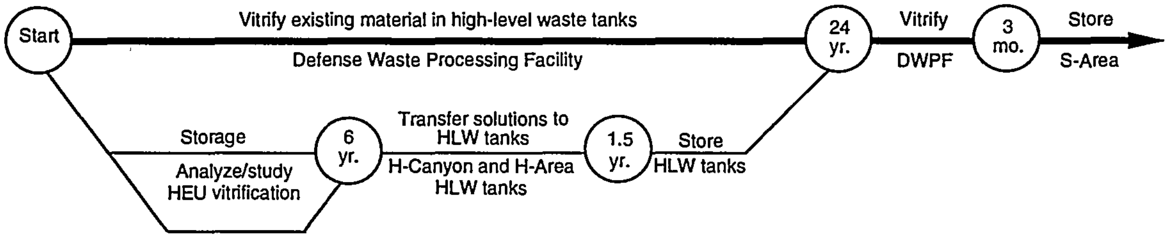
- **Processing to Oxide (Uranium Solidification Facility).**



TE DOE would continue to store the enriched uranium solution in H-Canyon while completing construction of the Uranium Solidification Facility (see Appendix C) in the canyon. After construction, DOE would use the H-Canyon process to remove radioactive decay products and other material from the solution and would transfer the solution to the Uranium Solidification Facility using intrafacility piping. DOE would process the solution to highly enriched uranium oxide, place the oxide in containers, and store the containers in a vault.

TE

- **Processing and Storage for Vitrification in the Defense Waste Processing Facility.**



TE DOE would continue to store the H-Canyon enriched uranium solution until it was ready for transfer to the H-Area high-level waste tanks. Before the transfer, DOE would adjust the solution to ensure the safety of the material already in the tanks. The material would be TE vitrified at the Defense Waste Processing Facility. Criticality concerns similar to those described in Section 2.3.1 would exist for this alternative.

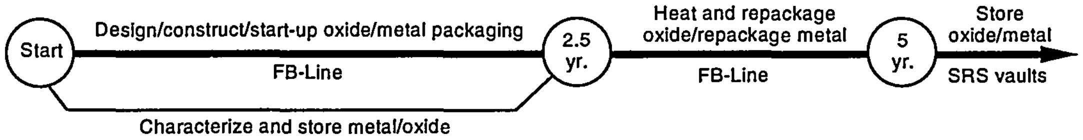
- **Continuing Storage (No Action).** DOE would continue to store the uranium solution in H-Canyon.

2.3.3 PLUTONIUM AND URANIUM STORED IN VAULTS

The material in this category is currently stored in about 3,000 containers, most of which are small cans in either the Building 235-F vault or the FB-Line vault. The material includes alloys, compounds, oxides, large metal pieces such as buttons and ingots, and metal fragments, and consists predominantly of plutonium-239 with some uranium-235 and neptunium-237. TC

DOE anticipates that the material would fall into one of two categories. The first would be material that DOE could stabilize to meet the long-term storage criteria (DOE 1994h) by simply heating and repackaging. TE The material in this category would generally be lower in chemical contaminants and higher in the percentage of fissile material; examples include plutonium metal (such as buttons) and plutonium and uranium oxides, which are essentially in product form. The other category of material would require some type of processing action to achieve stabilization. The material in this category would be higher in chemical contaminants (such as reactive calcium and fluorides) and lower in the percent of fissile material; examples include plutonium compounds, metal fragments, and plutonium and uranium oxides that are residual material from past production activities. DOE believes about half of all the containers hold material that would require only heating and repackaging; the remaining material would require a stabilization activity that involves processing.

- **Improving Storage.**



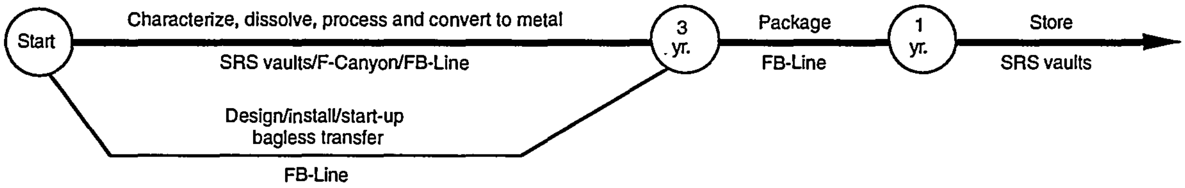
TC

DOE would upgrade its container inspection capability by installing new equipment in an existing facility such as FB-Line; this would consist of installing digital radiography screening equipment and other assay equipment to assess the condition of the material and the containers. DOE would transfer the containers to the inspection area to determine the condition of the material. Material determined to require processing before repackaging would be returned to storage until processing activities could be initiated. Material determined to require only repackaging would be returned to storage until the repackaging facility was completed.

In parallel with these inspection activities, DOE would begin work to provide the capability to meet the Departmental plutonium storage standard (DOE 1994h) in FB-Line. A glovebox TE would be added to heat plutonium oxide and to package oxide and metal in a nonreactive atmosphere without the use of plastic wrapping material. After the modifications were completed, DOE would transfer the plutonium oxide there for packaging. The packaged TC material would be placed in a SRS vault.

If DOE determined that it could not modify the FB-Line to provide the proper packaging capability or the capability for future inspection and packaging maintenance, DOE would TE begin work on the proposed Actinide Packaging and Storage Facility. This would be accomplished in parallel with plutonium inspection and characterization activities, but the facility would take about 8 years to complete and begin operations. Any plutonium oxide that had not been packaged to meet the DOE plutonium storage criteria (DOE 1994h) would be transferred to the facility and repackaged.

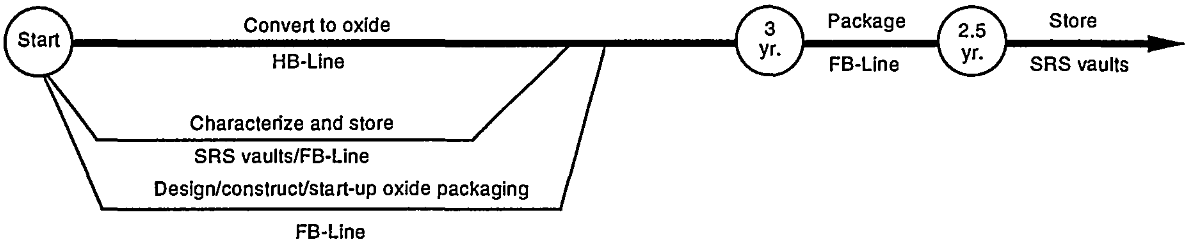
- **Processing to Metal.**



DOE would transfer potentially unstable oxide or metal from storage to F-Canyon or FB-Line, dissolve the material in one of the F-Canyon or FB-Line dissolvers, and process it as required TC in the canyon to separate the plutonium from the uranium and other impurities that contributed to the stability concerns. The plutonium would be processed through the FB-Line to produce plutonium metal, which would be packaged and placed in a vault for storage. No actions would occur to achieve a specific purity for this material other than those necessary to operate the process. The uranium would be processed to low enrichment by blending it with depleted uranium using FA-Line and F-Canyon process vessels or F-Area Outside Facilities tanks, as described for the H-Canyon uranium solutions (see Section 2.3.2). As a variation, the uranium could be chemically adjusted and transferred to the F-Area high-level waste tanks. The amount of fissile material involved in this transfer would be small, obviating the criticality concerns described for the Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative. In parallel with this effort, DOE would begin work to provide the capability to meet the Departmental plutonium storage standard (DOE 1994h) in FB-Line. A glovebox would be added or modified to package the material in a nonreactive atmosphere without the use of plastic wrapping material. After the modifications, DOE would transfer the plutonium metal there for packaging. The packaged material would be placed in an F-Area vault. High-level waste from these processing operations would be sent to the H-Area high-level waste tanks.

If DOE determined that it could not modify the FB-Line to provide the proper packaging capability or the capability for future inspection and packaging maintenance, it would begin work on the proposed Actinide Packaging and Storage Facility. This would be accomplished in parallel with plutonium conversion activities, but the facility would take about 8 years to complete and begin operations. Any plutonium metal that had not been packaged to meet the DOE plutonium storage criteria (DOE 1994h) would be transferred to the facility and repackaged.

- **Processing to Oxide.**



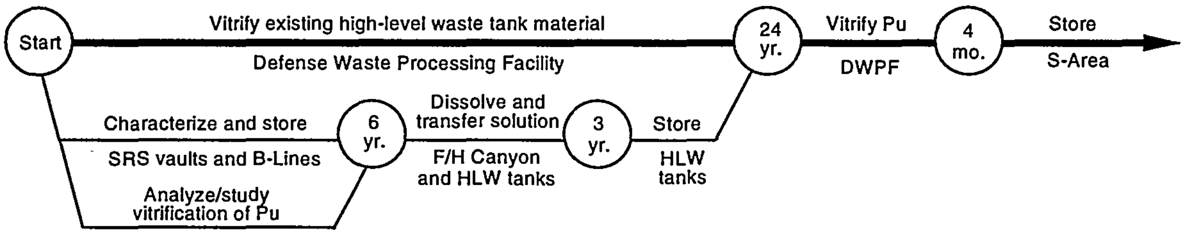
TC

DOE would transfer potentially unstable oxide or metal from storage to HB-Line or H-Canyon. In general, this alternative would be used to stabilize scrap plutonium-238 and material mixed with highly enriched uranium, and to increase overall efficiency through parallel operations with FB-Line. DOE would dissolve the material in one of the HB-Line or H-Canyon dissolvers and process it as required in the canyon to separate the plutonium from the uranium and other impurities that contributed to the stability concerns. The plutonium would be processed through HB-Line to produce an oxide, which would be placed in a vault for storage. No actions would occur to achieve a specific purity for this material other than those necessary to operate the process. The uranium would be diluted to low enrichment, converted to an oxide, and packaged as described for the H-Canyon uranium solutions (see Section 2.3.2). As a variation, the uranium could be chemically adjusted and transferred to the H-Area high-level waste tanks. The amount of fissile material involved in this transfer would be small, obviating the criticality concerns described for the Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative. In parallel with this effort, DOE would begin work to provide the capability to meet the Departmental plutonium storage standard (DOE 1994h) in FB-Line. A glovebox would be added or modified to heat and package the material in a nonreactive atmosphere without the use of plastic wrapping material. After the modifications, DOE would transfer the plutonium oxide there for packaging. The packaged material would be placed in an F-Area vault. High-level waste from these processing operations would be sent to the H-Area high-level waste tanks.

TC

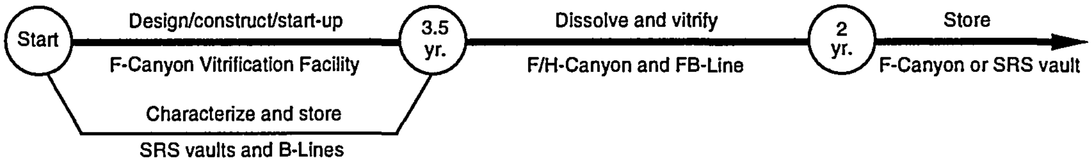
If DOE determined that it could not modify the FB-Line to provide the proper packaging capability or the capability for future inspection and packaging maintenance, it would begin work on the proposed Actinide Packaging and Storage Facility. This would be accomplished in parallel with oxide conversion activities, but the facility would take about 8 years to complete and begin operations. Any plutonium oxide that had not been packaged to meet the DOE plutonium storage criteria (DOE 1994h) would be transferred to the facility and repackaged.

- **Processing and Storage for Vitrification in the Defense Waste Processing Facility.**



DOE would store the material until it was ready to transfer it to the F- or H-Area high-level waste tanks. In preparing the material for transfer to the waste tanks, DOE would move it to FB-Line or F-Canyon or to HB-Line or H-Canyon and dissolve it. DOE would adjust the solution to ensure the safety of the material in the waste tanks and then would transfer the material to the F- or H-Area high-level waste tanks. The material would be vitrified at the Defense Waste Processing Facility. The difficulties associated with this alternative are the same as those described in Section 2.3.1

- **Vitrification (F-Canyon).**



DOE would store the potentially unstable oxide and metal until the proposed F-Canyon Vitrification Facility was available. Then the material would be transferred to F-Canyon or FB-Line and dissolved and processed in the canyon to separate the plutonium and uranium and other impurities. The plutonium would be chemically adjusted as required to achieve the feed specifications for vitrification and then vitrified. The resulting glass product in stainless-steel canisters would be stored in F-Canyon or a vault. The uranium would be processed to low enrichment by blending it with depleted uranium using FA-Line and F-Canyon process vessels or F-Area Outside Facilities tanks, as described in Section 2.3.2.

As a variation, the uranium could be chemically adjusted and transferred to the F-Area high-level waste tanks. The amount of fissile material involved in this transfer would be small, obviating the criticality concerns described for the Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative. Any high-level waste associated with this alternative would also be sent to the F-Area high-level waste tanks. TE

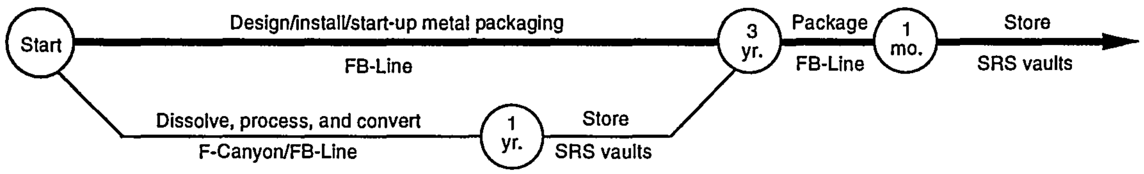
- **Continuing Storage (No Action).** DOE would continue to store the plutonium and uranium solids in a vault. TC

2.3.4 MARK-31 TARGETS

Approximately 16,000 metal targets are stored in the Receiving Basin for Offsite Fuel, the K- and L-Reactor disassembly basins, and the F-Canyon basin. These aluminum-clad targets contain depleted uranium, plutonium-239, and fission products. DOE has identified the following reasonable alternatives for the interim management of these targets. TC

- **Processing to Metal.**

TC



DOE would load the targets from the disassembly basins into large casks, load the casks on SRS rail cars, and transport them to F-Canyon, where it would load the targets in a dissolver tank and dissolve the targets. Then DOE would use the PUREX process to separate the plutonium solution from depleted uranium, fission products, and other impurities. DOE would process the depleted uranium to oxide in FA-Line and store it in F-Area, and would process the plutonium to metal in FB-Line. No actions would occur to achieve a specific purity for this material other than those necessary to operate the process. DOE would place the metal in containers and store the containers in a vault. In parallel with this effort, DOE would modify a portion of the existing FB-Line to provide the capability to package plutonium metal in a manner that met the storage criteria the Department has established for plutonium (DOE 1994h). A glovebox would be added to FB-Line to enable the metal to be packaged in a nonreactive atmosphere without the use of plastic wrapping material. On completing the modification to the FB-Line, DOE would repackage the material to meet the long-term storage criteria for plutonium metal.

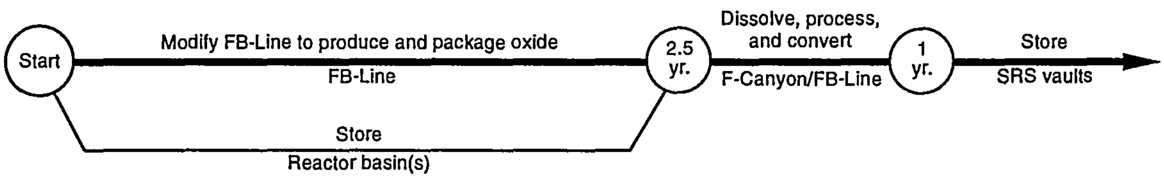
TC

If DOE determined that it could not modify the FB-Line to provide the proper packaging capability or the capability for future inspection and packaging maintenance, DOE would begin work on the proposed Actinide Packaging and Storage Facility; this would occur in parallel with plutonium conversion activities, but the facility would take about 8 years to complete and begin operations.

TE

- **Processing to Oxide.**

TE



DOE would load the targets from the disassembly basins into casks, load the casks on SRS rail cars, and transport them to F-Canyon, where it would dissolve the targets in a dissolver tank.

Then DOE would use the PUREX process to separate the plutonium solution from depleted uranium, fission products, and other impurities. DOE would modify the FB-Line to support conversion of the plutonium solutions to plutonium oxide and to package the material for storage. No actions would occur to achieve a specific purity for the material other than those necessary to operate the process. DOE would produce a material form and packaging configuration that met the DOE standard for long-term storage of plutonium oxide (DOE 1994h). DOE would process the depleted uranium to an oxide in FA-Line and store the material in F-Area. Any high-level waste from these processing activities would be transferred to the F-Area high-level waste tanks. TE

If the extent of the FB-Line modifications necessary to meet the DOE plutonium storage standard were economically or physically impractical (i.e., too expensive or not enough space for the equipment required), the Department would perform the stabilization effort in two phases. DOE would convert the material initially to an oxide form and package it in FB-Line. In parallel, DOE would construct the proposed Actinide Packaging and Storage Facility. The oxide initially produced would be stored in a vault until the new facility was available. DOE estimates that the minimum required modifications to FB-Line would take about 2-1/2 years to complete and that the Actinide Packaging and Storage Facility would be available in approximately 8 years. TC

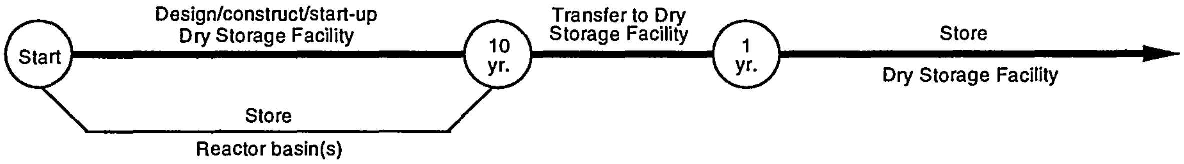
DOE considered two other variations of this alternative. DOE could dissolve the Mark-31 targets in H-Canyon and process the resulting plutonium solutions into an oxide in HB-Line. This variation would require modification of the HB-Line to provide the capability to package the resulting oxide in accordance with the DOE standard for long-term storage of plutonium (DOE 1994h). Approximately 3 years would be required to make the necessary modifications. However, even if DOE modified HB-Line, the volume of depleted uranium contained in the Mark-31 targets would require the operation of H-Canyon for more than 30 years. TE

As another variation, DOE could dissolve the Mark-31 targets in F-Canyon, transport the resulting plutonium solutions to H-Canyon, and convert the plutonium to an oxide using HB-Line. Approximately 1 year would be required to modify the H-Canyon and F-Canyon facilities to provide the capability to load and unload the solutions into a transport container. DOE does not currently have a container designed to transport liquid plutonium, but is exploring the availability of such a container internationally. As in the variation described above, approximately 3 years would be required to modify HB-Line to provide the capability

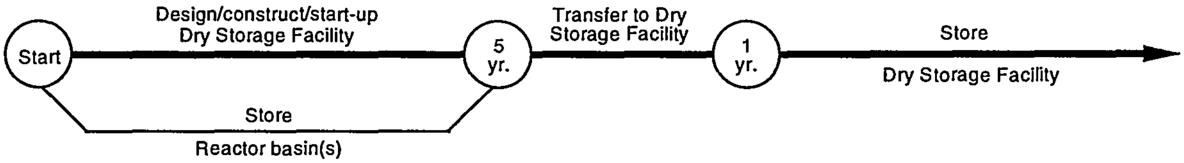
TC to package the oxide in accordance with the DOE standard. It would take more than 6 years to convert the solutions to an oxide in HB-Line, as opposed to approximately 1 year in a modified FB-Line with the same capability. Some of the necessary facility modifications and dissolution operations could take place in parallel. However, even if DOE can find or develop a container suitable for transport of the plutonium solutions, the total time required to convert and package the plutonium contained in the Mark-31 targets into an oxide using this variation would be more than 9 years (as opposed to 4 years using a modified FB-Line). For these reasons, DOE did not consider these two variations to be reasonable alternatives warranting detailed analysis.

- **Improving Storage.** In addition to the 10-year period evaluated in the Draft EIS for the implementation of this alternative for the Mark-31 targets, DOE has evaluated an accelerated (5-year) implementation schedule.

Traditional Dry Storage Facility Design and Construction Schedule



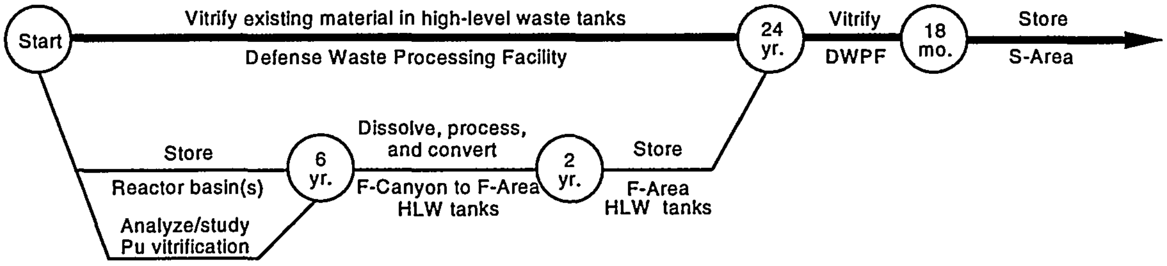
L11-9 **Accelerated Dry Storage Facility Design and Construction Schedule**



Under this alternative DOE would place the targets in dry storage. Because of technical uncertainties (e.g., potential pyrophoricity, elimination of potential reactive material) associated with the dry storage of failed fuel, DOE would perform additional research to demonstrate technical concepts for drying and placing aluminum-clad fuels and targets with failed cladding into canisters for storage. Some work related to the dry storage of spent nuclear fuel has already been done in the United States and other countries. DOE would expand this body of information through research and laboratory-scale development work for the specific types of fuel and targets stored at the SRS. In conjunction with this work, DOE would design the equipment and facility features required for dry storage. DOE has estimated that the development, design, and construction of a dry storage facility could be completed in between 5 and 10 years. A traditional design and construction schedule similar to that used on other major nuclear facilities could require as long as 10 years. DOE estimates that it could design and construct a facility in about 5 years if certain actions occurred to accelerate the schedule (e.g., a request for emergency funding from Congress and establishment of a line item project in the 1998 fiscal budget for DOE). The accelerated design and construction schedule would include an integrated approach for procurement of design and construction services to DOE (i.e., a "turnkey" job from a single vendor). A typical dry storage facility would be a Modular Dry Storage Vault (see Appendix C). This facility would consist of four major components: a receiving/unloading area, fuel storage canisters, a shielded container handling machine, and a modular vault for storing the fuel in storage canisters. As a variation, canisters could be stored in dry storage casks rather than a vault. DOE anticipates that the time and costs to construct such a facility using either vault or cask storage would be similar (DOE 1995e). L11-9

A process to place the Mark-31 targets in dry storage would be to (1) remove the targets from the basins and drain and dry them; (2) can the targets or place them directly in canisters, and back-fill the cans or canisters with an inert gas to inhibit further corrosion; and (3) if cans were used, load them in storage canisters. This process could be varied as dictated by the condition of the material. After the targets were loaded in a canister, a machine would transport the canister to the modular storage vault. The vault would consist of a large concrete structure with an array of vertical tubes to hold the canisters. The canister transport machine would move into the vault and load the canister in to a storage tube. A shielded plug would be placed on top of the tube. The transport machine and the vault storage tubes would be heavily shielded to reduce radiation levels from the canister. To use dry storage casks, the machine would transport the canister to a cask (horizontal or vertical) and discharge the canister into the cask; then the cask would be sealed. DOE estimates that it would need approximately 1 year to place the Mark-31 targets in dry storage. Until the Dry Storage Facility was completed, DOE would continue to store Mark-31 targets in existing basins, as described below for No Action.

- **Processing and Storage for Vitrification in the Defense Waste Processing Facility.** TE

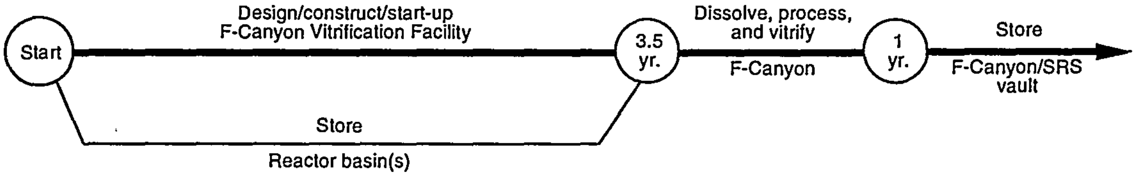


TC

DOE would continue to store the Mark-31 targets until it was ready to transfer material to the high-level radioactive waste system. DOE would process the existing depleted uranium solutions in F-Canyon through the FA-Line to make room for processing the Mark-31 targets. The resulting depleted uranium oxide would be loaded in 208-liter (55-gallon) drums and placed in storage. In F-Canyon, DOE would dissolve the targets and then process the material to separate the plutonium from the depleted uranium. Then, rather than transferring the plutonium solution to FB-Line, DOE would add chemicals to preclude the potential for a nuclear criticality, concentrate and neutralize the solution, and discharge the mixture to the F-Area high-level waste tanks. DOE would vitrify the material at the Defense Waste Processing Facility; the difficulties associated with this process would be the same as those described in Section 2.3.1 for the H-Canyon plutonium solutions. The depleted uranium would be converted to an oxide in FA-Line, packaged, and placed in storage. The high-level waste generated during the chemical separation and chemical adjustment operations would be sent to the F-Area high-level waste tanks. TE

TC As a variation to this alternative, after the completion of technical studies DOE could dissolve the Mark-31 targets and transfer all of the material to the high-level waste tanks. The depleted uranium would not be recovered but would be sent to the high-level waste tanks with the other material.

- **Vitrification (F-Canyon).**



DOE could use the proposed F-Canyon Vitrification Facility to vitrify the plutonium in the Mark-31 targets. DOE would continue to store the material until the new facility was available. Then the material would be transferred to F-Canyon and dissolved. The material would be processed to separate the depleted uranium from the plutonium, and the plutonium would be vitrified. The depleted uranium solution would be converted to depleted uranium oxide in FA-Line. Any high-level waste from these operations would be transferred to the F-Area high-level waste tanks.

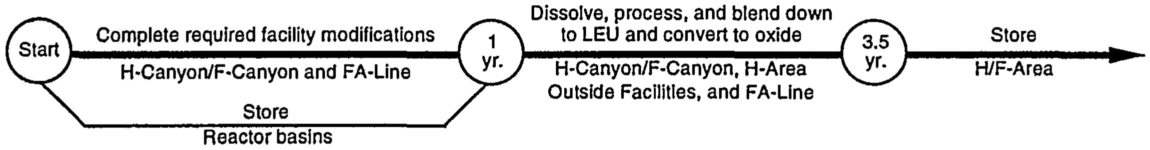
TC
- **Continuing Storage (No Action).** DOE would continue to store the Mark-31 targets in the water-filled basins.

As a precursor to the Processing to Metal, Processing to Oxide, Processing and Storage for Vitrification in the Defense Waste Processing Facility, and the Vitrification (F-Canyon) Alternatives, DOE could dissolve unirradiated depleted uranium targets (which would result in no fissile material or fission products) in the F-Canyon dissolvers as part of equipment testing and operator training evaluations. TE

2.3.5 MARK-16 AND -22 FUELS

Approximately 1,900 irradiated fuel assemblies are stored in water-filled basins in the K-, L- and P-Reactor areas and in the H-Canyon facility. The fuel tubes contain highly enriched uranium and are clad in aluminum. DOE has identified the following alternatives for management of these fuels:

- **Blending Down to Low Enriched Uranium.**



DOE would load the fuel tubes from the disassembly basins into casks, transport the casks to H-Canyon, dissolve the fuels, and separate enriched uranium from fission products, neptunium, and the small quantities of plutonium normally found in the fuel. This would be accomplished using the normal H-Canyon process. The fission products, plutonium, and other impurities would be transferred to the H-Area high-level waste tanks. The enriched uranium would be blended with depleted uranium and stabilized, as described in Section 2.3.2. TE

If DOE selected this alternative for the uranium solutions in H-Canyon and the Mark-16 and Mark-22 fuel, it would build only one storage facility for low enriched uranium oxide, which would have an area of about 557 square meters (6,000 square feet). TC

As a variation to this alternative, DOE could transport the fuel to F-Canyon for processing. In this case, the blending operations would occur immediately after the fuel dissolving operations. Depleted uranium from FA-Line or from material already in the canyon would be added after the dissolution process. The SRS has sufficient stores of depleted uranium to support any required blending operation (see Table 1-1). The resulting low enriched uranium would be separated from the other material and radioactive decay products in the fuel and transferred to FA-Line for conversion to uranium oxide. The oxide would be stored in 208-liter (55-gallon) TC drums. The fission products and other materials would be transferred to the F-Area high-level waste tanks.

- **Processing to Oxide (Uranium Solidification Facility).**

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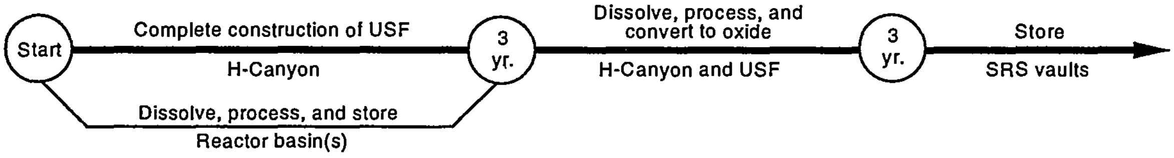
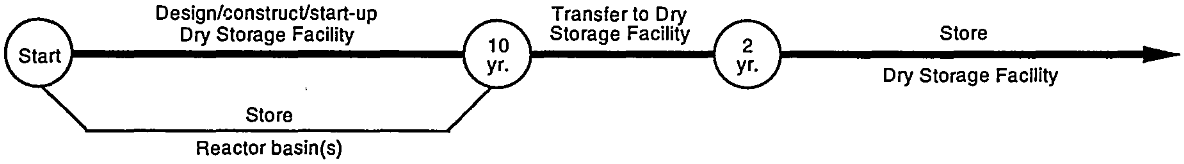


DOE would continue to store the fuel while completing construction of the Uranium Solidification Facility in H-Canyon. After construction, DOE would process the fuel as described in Section 2.3.2, transfer the resulting enriched uranium solution to the Uranium Solidification Facility, convert the uranium solution to an oxide, package the oxide, and place the containers in a vault for storage.

- **Improving Storage.**

Traditional Dry Storage Facility Design and Construction Schedule



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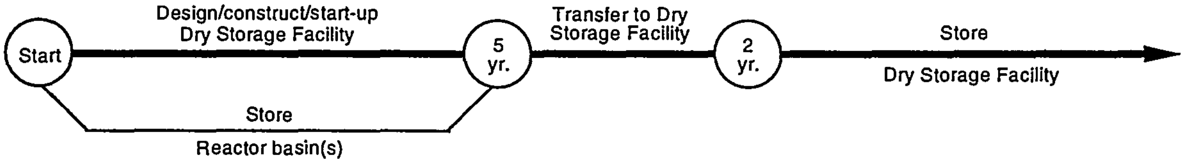
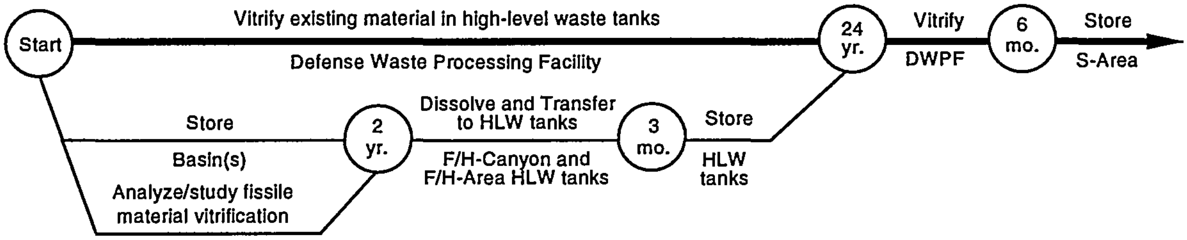
Accelerated Dry Storage Facility Design and Construction Schedule



Under this alternative DOE would place the spent fuel in dry storage using an approach similar to that described above for the Mark-31 targets. Because of technical uncertainties associated with dry-storing failed fuel, DOE would perform research and development work to demonstrate the technical concepts for storing the Mark-16 and -22 fuel in dry conditions. However, DOE recognizes the preparation of failed metal-alloy fuel for dry storage could be simpler than the preparation of failed targets for dry storage. DOE would design and construct the same dry storage facility as that described above for the Mark-31 targets. The only differences would be in the design of the handling and drying equipment to accommodate the physical differences between the Mark-31 targets (depleted uranium metal slugs) and the Mark-16 and -22 fuel (long highly enriched uranium-aluminum alloy tubes). DOE estimates that the design and construction schedules would be the same as those for the Dry Storage Facility described for the Mark-31 targets. A traditional design and construction schedule could take as long as 10 years, and an accelerated schedule could take about 5 years. The size and technical requirements for the Dry Storage Facility would be essentially the same as those described for the Mark-31 targets. DOE estimates that it would take approximately 2 years to place the material in dry storage. Until the Dry Storage Facility was complete, DOE would continue to store the Mark-16 and -22 fuel in the existing basins, as described below under No Action. The process to place the fuel into dry storage would be essentially the same as that described for the Mark-31 targets.

TC

- **Processing and Storage for Vitrification in the Defense Waste Processing Facility.**



TE

DOE would continue to store the material in solid form until it could complete technical studies on the transfer of fissile solutions to the high-level waste tanks. When the studies were complete, DOE would move the material to H-Canyon and dissolve it, adjust the resulting solution to ensure the nuclear criticality safety of the material in the waste tanks, and vitrify the material at the Defense Waste Processing Facility. The difficulties associated with this process would be the same as those described in Section 2.3.1 for the H-Canyon plutonium solutions.

TC

- **Continuing Storage (No Action).** DOE would continue to store the Mark-16 and -22 fuel in a water-filled basin. Compensatory actions would continue as described for the activities included under no action (see the introductory portion of Chapter 2).

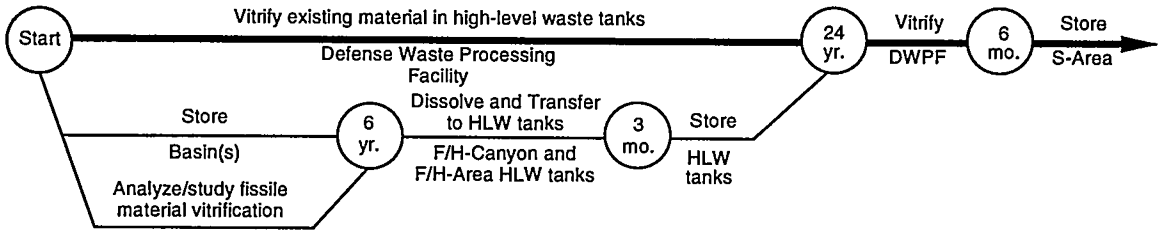
DOE did not evaluate Processing to Metal because this capability does not exist at the SRS and, because the oxide form of the material would be stable, there would be no advantage in developing the capability to produce uranium metal.

2.3.6 OTHER ALUMINUM-CLAD TARGETS

Approximately 1,800 metal target elements are stored in water-filled basins in the K-, L-, and P-Areas. These elements contain small amounts of fissile material; primarily they contain such materials as thorium, cobalt, and thulium. DOE has identified the following alternatives for management of these targets:

- **Processing and Storage for Vitrification in the Defense Waste Processing Facility.**

TC

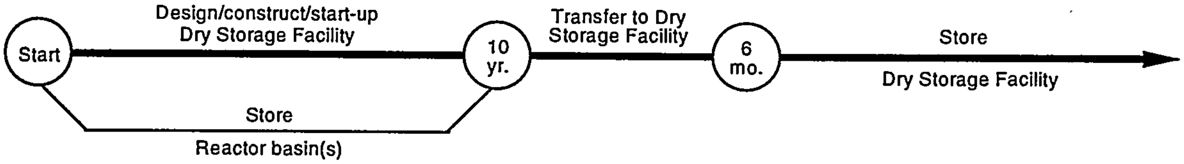


DOE would continue to store the material in its current form until it could complete technical studies on the transfer of fissile solutions to the high-level waste tanks. DOE anticipates that these studies would be simpler than those for other material evaluated in this EIS because the fissile material content of these items is relatively low. When the studies were complete, DOE would move the material to a B-Line or canyon and dissolve it. DOE would adjust the resulting solution to ensure the safety of the material in the waste tanks from nuclear criticality. The material would be vitrified at the Defense Waste Processing Facility.

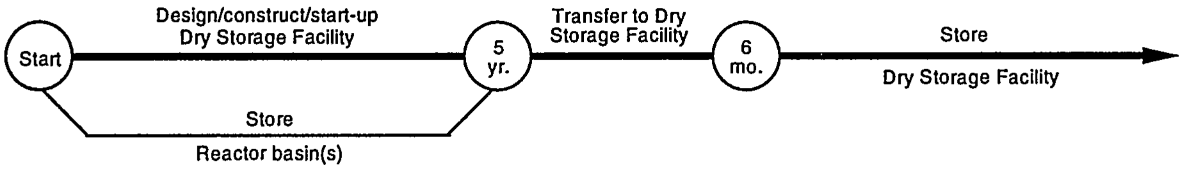
- **Improving Storage.**

Traditional Dry Storage Facility Design and Construction Schedule

TC



Accelerated Dry Storage Facility Design and Construction Schedule



DOE would use the same techniques, facility, and equipment to place the other aluminum-clad targets in dry storage as those described for the Improving Storage Alternative for Mark-31 targets and Mark-16 and -22 fuel (see Sections 2.3.4 and 2.3.5, respectively). Research and development work would demonstrate the technical concepts for storing the other aluminum-clad targets in dry conditions. DOE would design and construct the same Dry Storage Facility as that described above for the Mark-31 targets. The only differences could be in the design of the handling and drying equipment to accommodate the physical differences between the Mark-31 targets (depleted uranium metal slugs) and the other aluminum-clad targets. DOE TC estimates that the design and construction schedules would be the same as those for the Dry Storage Facility described above for the Mark-31 targets. A traditional design and construction schedule could take as long as 10 years and an accelerated schedule could take about 5 years. The size and technical requirements for the Dry Storage Facility would be essentially the same as those described for the Mark-31 targets. DOE estimates that it would take approximately 1 year to place the material in dry storage. Until the Dry Storage Facility was complete, DOE would continue to store other aluminum-clad targets in the existing basins, as described below under No Action. The process to place the targets into dry storage would be essentially the same as that described for the Mark-31 targets.

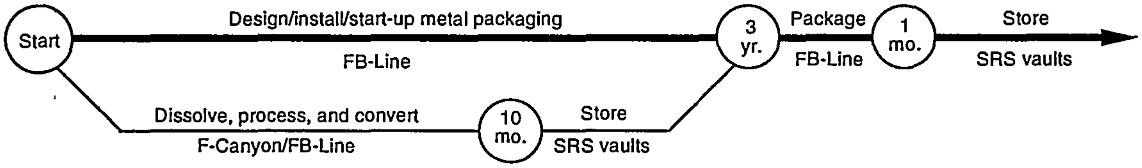
- **Continuing Storage (No Action).** DOE would continue to store the targets in a water-filled basin. Compensatory actions would continue as described for the activities included under no action (see the introductory portion of Chapter 2).

2.3.7 ALUMINUM-CLAD TAIWAN RESEARCH REACTOR FUEL AND EXPERIMENTAL BREEDER REACTOR-II SLUGS

L3-1
L4-1
L5-5

This material consists of 81 canisters of failed natural uranium Taiwan Research Reactor (TRR) fuel and one canister of Experimental Breeder Reactor-II (EBR-II) depleted uranium slugs that has failed and is venting gas and possibly radionuclides to the basin. These canisters are stored in the Receiving Basin for Offsite Fuel. The TRR fuel is clad with aluminum, but the cladding was damaged as a result of storage conditions in Taiwan and could not be relied on to contain the fuel and fission products. The fuel was shipped to the SRS in aluminum cans, one damaged fuel rod per can. The EBR-II fuel was originally clad with stainless steel but the cladding was removed before the material was shipped because the SRS chemical separations facilities are not configured to process material with stainless-steel cladding. After the cladding was removed, the EBR-II material was loaded in a can for shipment and subsequent wet storage. DOE evaluated the following alternatives for stabilization of the TRR fuel and EBR-II slugs.

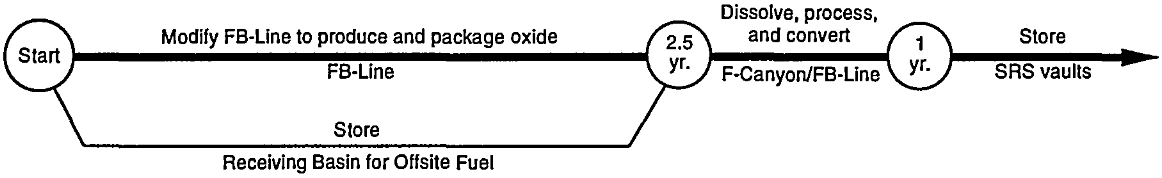
- **Processing to Metal.**



DOE would load the canisters of failed TRR fuel and declad EBR-II slugs into large casks, load the casks on SRS rail cars, and transport them to F-Canyon, where it would load the canisters in a dissolver tank and dissolve the fuel and slugs. Then DOE would use the PUREX process to separate the plutonium solution from depleted uranium, fission products, and other impurities. DOE would process the depleted uranium to oxide in FA-Line and store it in F-Area, and would process the plutonium to metal in FB-Line. No actions would occur to achieve a specific purity for this material other than those necessary to operate the process. DOE would place the metal in containers and store the containers in a vault. In parallel with this effort, DOE would modify a portion of the existing FB-Line to provide the capability to package plutonium metal in a manner that met the storage criteria the Department has established for plutonium (DOE 1994h). A glovebox would be added to FB-Line to enable the metal to be packaged in a nonreactive atmosphere without the use of plastic wrapping material. On completing the modification to the FB-Line, DOE would repackage the material to meet the long-term storage criteria for plutonium metal. L3-1 L4-1 L5-5

If DOE determined that it could not modify the FB-Line to provide the proper packaging capability or the capability for future inspection and packaging maintenance, DOE would begin work on the proposed Actinide Packaging and Storage Facility; this would occur in parallel with plutonium conversion activities, but the facility would take about 8 years to complete and begin operations.

- **Processing to Oxide.**



DOE would load the canisters of failed TRR fuel and declad EBR-II slugs into casks, load the casks on SRS rail cars, and transport them to F-Canyon, where it would dissolve the targets in a dissolver tank. Then DOE would use the PUREX process to separate the plutonium solution from depleted uranium, fission products, and other impurities. DOE would modify the FB-Line to support conversion of the plutonium solutions to plutonium oxide and to package the material for storage. No actions would occur to achieve a specific purity for the material other than those necessary to operate the process. DOE would produce a material form and packaging configuration that met the DOE standard for long-term storage of plutonium oxide (DOE 1994h). DOE would process the depleted uranium to an oxide in FA-Line and store the material in F-Area. Any high-level waste from these processing activities would be transferred to the F-Area high-level waste tanks.

L3-1
L4-1
L5-5

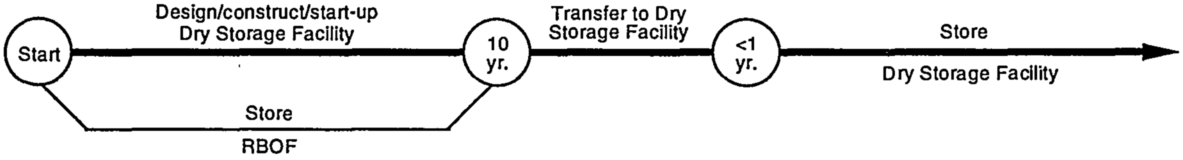
If the extent of the FB-Line modifications necessary to meet the DOE plutonium storage standard were economically or physically impractical (i.e., too expensive or not enough space for the equipment required), the Department would perform the stabilization effort in two phases. DOE would convert the material initially to an oxide form and package it in FB-Line. In parallel, DOE would construct the proposed Actinide Packaging and Storage Facility. The oxide initially produced would be stored in a vault until the new facility was available. DOE estimates that the minimum required modifications to FB-Line would take about 2-1/2 years to complete and that the Actinide Packaging and Storage Facility would be available in approximately 8 years.

DOE considered two other variations of this alternative. DOE could dissolve the TRR fuel and EBR-II slugs targets in H-Canyon and process the resulting plutonium solutions into an oxide in HB-Line. This variation would require modification of the HB-Line to provide the capability to package the resulting oxide in accordance with the DOE standard for long-term storage of plutonium (DOE 1994h). Approximately 3 years would be required to make the necessary modifications.

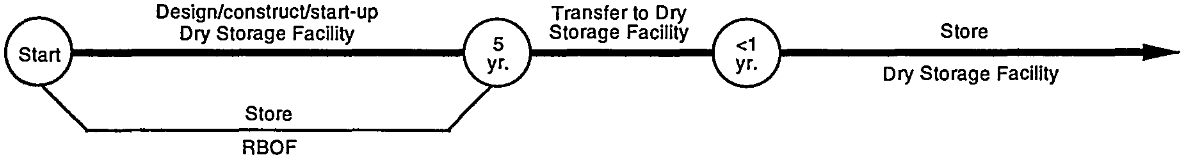
As another variation, DOE could dissolve the TRR fuel and EBR-II slugs targets in F-Canyon, transport the resulting plutonium solutions to H-Canyon, and convert the plutonium to an oxide using HB-Line. Approximately 1 year would be required to modify the H-Canyon and F-Canyon facilities to provide the capability to load and unload the solutions into a transport container. DOE does not currently have a container designed to transport liquid plutonium, but is exploring the availability of such a container internationally. As in the variation described above, approximately 2-1/2 years would be required to modify HB-Line to provide the capability to package the oxide in accordance with the DOE standard. It would take more than 6 years to convert the solutions to an oxide in HB-Line, as opposed to approximately 1 year in a modified FB-Line with the same capability. Some of the necessary facility modifications and dissolution operations could take place in parallel. However, even if DOE can find or develop a container suitable for transport of the plutonium solutions, the total time required to convert and package the plutonium contained in the TRR fuel and EBR-II slugs into an oxide using this variation would be more than 9 years (as opposed to less than 4 years using a modified FB-Line). For these reasons, DOE did not consider these two variations to be reasonable alternatives warranting detailed analysis.

L3-1
L4-1
L5-5
- **Improving Storage.** DOE has evaluated two schedules for the implementation of this alternative for the TRR fuel and EBR-II slugs, a traditional (10-year) schedule and an accelerated (5-year) schedule.

Traditional Dry Storage Facility Design and Construction Schedule

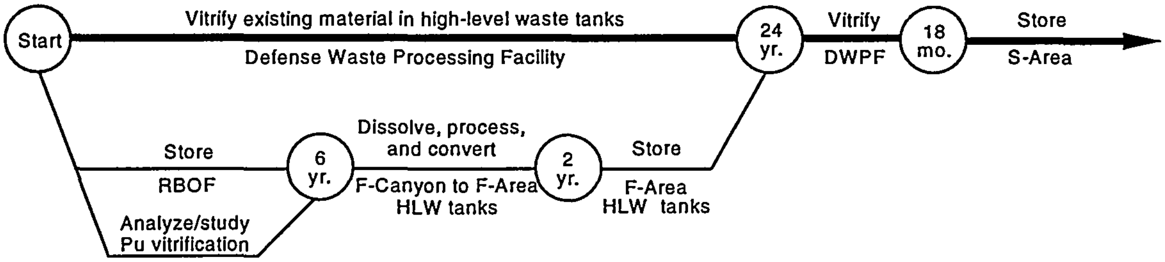


Accelerated Dry Storage Facility Design and Construction Schedule



Under this alternative, DOE would place the TRR and EBR-II material in dry storage. Research and development work would demonstrate the technical concepts for storing the TRR and EBR-II material in dry conditions. DOE would design and construct the same Dry Storage Facility as that described above for the Mark-31 targets. The only differences could be in the design of the handling and drying equipment to accommodate the physical differences between the Mark-31 targets (depleted uranium metal slugs) and the TRR and EBR-II material. DOE estimates that the design and construction schedules would be the same as those for the Dry Storage Facility described above for the Mark-31 targets. A traditional design and construction schedule could take as long as 10 years and an accelerated schedule could take about 5 years. The size and technical requirements for the Dry Storage Facility would be essentially the same as those described for the Mark-31 targets. DOE estimates that it would take less than 1 year to place the material in dry storage. Until the Dry Storage Facility was complete, DOE would continue to store the TRR and EBR-II material in the existing basins, as described below under No Action. DOE believes that only the modular cask dry storage method would be practical for the small quantity of TRR and EBR-II material evaluated. The vault storage method could be viable if implemented in conjunction with larger volumes of material in another category, such as Mark-31 targets or Mark-16 or -22 fuel.

- **Processing and Storage for Vitrification in the Defense Waste Processing Facility.**



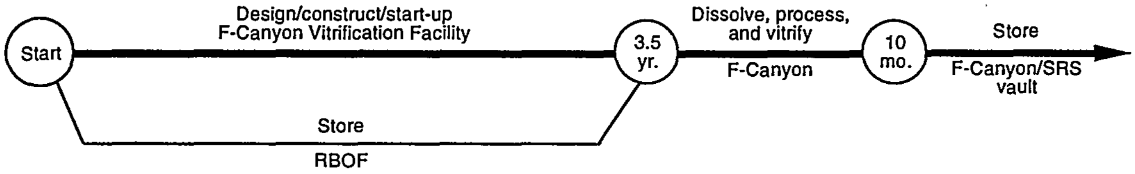
L3-1
L4-1
L5-5

DOE would continue to store the failed TRR fuel and EBR-II slugs until it was ready to transfer material to the high-level radioactive waste system. DOE would process the existing depleted uranium solutions in F-Canyon through the FA-Line to make room for processing the TRR fuel and EBR-II slugs, if necessary. The resulting depleted uranium oxide would be loaded in 208-liter (55-gallon) drums and placed in storage. In F-Canyon, DOE would dissolve the canisters of fuel and slugs and then process the material to separate the plutonium from the depleted uranium. Then, rather than transferring the plutonium solution to FB-Line, DOE would add chemicals to preclude the potential for a nuclear criticality, concentrate and neutralize the solution, and discharge the mixture to the F-Area high-level waste tanks. DOE would vitrify the material at the Defense Waste Processing Facility; the difficulties associated with this process would be the same as those described in Section 2.3.1 for the H-Canyon plutonium solutions. The depleted uranium would be converted to an oxide in FA-Line, packaged, and placed in storage. The high-level waste generated during the chemical separation and chemical adjustment operations would be sent to the F-Area high-level waste tanks.

As a variation to this alternative, after the completion of technical studies DOE could dissolve the canisters of fuel and slugs and transfer all of the material to the high-level waste tanks. The depleted uranium would not be recovered but would be sent to the high-level waste tanks with the other material. This variation could be completed sooner than the approach described above, but would likely involve a greater volume of waste being transferred and eventually vitrified in DWPF.

- **Vitrification (F-Canyon).**

L3-1
L4-1
L5-5



DOE could use the proposed F-Canyon Vitrification Facility (see Appendix C) to vitrify the plutonium in the TRR fuel and EBR-II slugs. DOE would continue to store the material in RBOF until the new facility was available. Then the material would be transferred to F-Canyon and dissolved. The material would be processed to separate the depleted uranium from the plutonium, and the plutonium would be vitrified. The depleted uranium solution would be converted to depleted uranium oxide in FA-Line. Any high-level waste from these operations would be transferred to the F-Area high-level waste tanks.

- **Continuing Storage (No Action).** DOE would continue to store the canisters of TRR fuel and EBR-II slugs in the RBOF. DOE would place the failed canisters inside an overpack designed to contain corrosion products and insoluble fission products. The overpack would be vented and a system would be designed for collection and treatment of gas evolving from the failed canisters.

2.4 Comparison of Environmental Impacts Among the Alternatives

Tables 2-2 through 2-12 compare environmental impacts estimated to occur over the next 10 years for each of the alternatives described in Sections 2.2 and 2.3. This section summarizes the environmental impacts for each of the factors contained in the tables. These tables also list estimated impacts (latent cancer fatalities) from the potential accident with the highest consequences associated with each alternative. Chapter 4 also contains tables of 10-year impacts, but these tables include additional parameters. For example, Tables 2-2 through 2-12 list the incremental contribution each alternative could make to the highest annual concentration of nitrogen oxide in the air, measured at the SRS boundary. The Chapter 4 tables list the highest 1-hour, 3-hour, 8-hour, 12-hour, 24-hour, weekly, or annual average concentration of the pollutants of concern in the air around the SRS. Tables 2-2 through 2-12 list only nitrogen oxide, because it is typically the nonradiological air pollutant of primary interest. In other words, the 10-year impact data listed in these tables are a subset of the more detailed 10-year impact data presented in Chapter 4.

To forecast the potential impacts of the implementation of each alternative over the next 10 years, DOE identified the various steps of "phases" required. DOE anticipates that the level of environmental impacts would depend on the types of activities to be performed to manage the nuclear materials. For example, DOE expects different impacts from processing or repackaging operations than from maintaining nuclear material in a storage vault. L11-1

Appendix D contains annual estimates for the phases of each alternative. DOE used the data in Appendix D along with a schedule of potential activities to estimate the 10-year impacts for each alternative. Similarly, Appendix E analyzes environmental impacts from potential accidents associated with the phases of each alternative. The tables in Appendix E identify the accidents that pose either the greatest risk (considering both the likelihood and the consequences) or the greatest consequence (in latent cancer fatalities) in **bold** type. Again, DOE included the highest consequence accident for each alternative.

2.4.1 HEALTH EFFECTS

As indicated in Tables 2-2 through 2-12, the radiological health effects from normal operations (including transportation activities) would vary among the alternatives, but all would result in less than one additional latent cancer fatality in the population surrounding the SRS and in the worker population over the 10-year period. The health effects from potential facility or transportation accidents involving the alternatives range from less than 1 to 38 additional latent cancer fatalities in

Table 2-2. Comparison of the potential environmental impacts of the alternatives for stable material.[a]

TE

Factors	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)	Vitrification (F-Canyon)	Improving Storage
	Alternatives						
Health effects of Normal Operations							
Radiological health effects (10-year totals):							
Offsite population latent cancer fatalities	0.0006	NA[b]	NA	NA	NA	NA	NA
Worker latent cancer fatalities	0.056	NA	NA	NA	NA	NA	NA
Health effects to offsite population from facility accidents[c] (projected latent cancer fatalities)	0.48	NA	NA	NA	NA	NA	NA
Risk (latent cancer fatalities per year)[d]	1.2×10^{-7}	NA	NA	NA	NA	NA	NA
Health effects from transportation (projected latent cancer fatalities)							
Incident-free (involved worker)	0.011	NA	NA	NA	NA	NA	NA
Accidents (offsite population)[e]	0.0000021	NA	NA	NA	NA	NA	NA
Air resources							
Nonradiological - Nitrogen oxide incremental concentration at SRS boundary (highest annual, micrograms per cubic meter)	0.0053	NA	NA	NA	NA	NA	NA
Water resources							
Lead (micrograms per liter) discharged to SRS streams	0	NA	NA	NA	NA	NA	NA
Utilities (10-year totals)							
Electricity usage (megawatt-hour)	400,000	NA	NA	NA	NA	NA	NA
Waste management (10-year totals)							
High-level liquid waste generation (million liters)	21	NA	NA	NA	NA	NA	NA
Equivalent DWPF canisters	40	NA	NA	NA	NA	NA	NA
Saltstone generation (cubic meters)	11,000	NA	NA	NA	NA	NA	NA
Transuranic waste generation (cubic meters)	20	NA	NA	NA	NA	NA	NA
Hazardous/mixed waste generation (cubic meters)	60	NA	NA	NA	NA	NA	NA
Low-level radioactive waste generation (cubic meters)	41,000	NA	NA	NA	NA	NA	NA

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a. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.
b. NA = Not applicable.
c. Assumes highly unlikely occurrence of maximum consequence accident.
d. Highest point estimate of risk during any phase of the alternative. The risk is calculated as the potential consequence of an accident (in latent cancer fatalities) multiplied by the probability per year of the accident occurring.
e. Maximum reasonably foreseeable latent cancer fatalities from medium probability accident based on the shipment of waste.

TE

Table 2-3. Comparison of the potential environmental impacts of the alternatives for plutonium-242.[a] TE

Factors	Alternatives: Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[b]	Vitrification (F-Canyon)	Improving Storage
Health effects of Normal Operations							
Radiological health effects (10-year totals):							
Offsite population latent cancer fatalities	0.00025	0.00024	0.0017	NA[c]	0.00016	0.0017	NA
Worker latent cancer fatalities	0.0052	0.035	0.024	NA	0.0035	0.027	NA
Health effects to offsite population from facility accidents[d] (projected latent cancer fatalities)	6.5	6.5	6.5	NA	38	6.5	NA
Risk (latent cancer fatalities per year)[e]	0.000015	0.000015	0.000015	NA	0.000015	0.000015	NA
Health effects from transportation[f] (projected latent cancer fatalities)							
Incident-free (involved worker)	0.0017[g]	0.0019	0.0011	NA	0.0011[g]	0.0012	NA
Accidents (offsite population)[h]	0.0000021	2	2	NA	0.0000021	2	NA
Air resources							
Nonradiological - Nitrogen oxide incremental concentration at SRS boundary (highest annual, micrograms per cubic meter)	0.012	0.140	0.033	NA	0.033	0.11	NA
Water resources							
Lead (micrograms per liter) discharged to SRS streams	3.2	2.7	2.7	NA	3.2	2.8	NA
Utilities (10-year totals)							
Electricity usage (megawatt-hour)	133,000	127,000	41,000	NA	85,000	42,000	NA
Waste management (10-year totals)							
High-level liquid waste generation (million liters)	1.2	1.2	0.12	NA	0.77	0.16	NA
Equivalent DWPF canisters	20	21	2	NA	23	3	NA
Saltstone generation (cubic meters)	3,300	3,200	330	NA	2,100	420	NA
Transuranic waste generation (cubic meters)	0	21	56	NA	0	61	NA
Hazardous/mixed waste generation (cubic meters)	0	0	60	NA	0	60	NA
Low-level radioactive waste generation (cubic meters)	5,600	6,500	4,300	NA	3,500	4,700	NA

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a. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.
b. The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.
c. NA = Not applicable.
d. Assumes highly unlikely occurrence of maximum consequence accident.
e. Highest point estimate of risk during any phase of the alternative. The risk is calculated as the potential consequence of an accident (in latent cancer fatalities) multiplied by the probability per year of the accident occurring.
f. Includes transportation of associated radioactive waste.
g. Waste transportation only.
h. Maximum reasonably foreseeable latent cancer fatalities from medium probability accident based on the shipment of waste.

TE

Table 2-4. Comparison of the potential environmental impacts of the alternatives for americium and curium.[a,b]

TE

Factors	Alternatives						
	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[c]	Vitrification (F-Canyon)	Improving Storage
Health effects of Normal Operations							
Radiological health effects (10-year totals):							
Offsite population latent cancer fatalities	0.00035	NA[d]	0.0012	NA	0.00041	0.00050	NA
Worker latent cancer fatalities	0.034	NA	0.128	NA	0.044	0.052	NA
Health effects to offsite population from facility accidents[e] (projected latent cancer fatalities)	3.1	NA	6.5	NA	38	6.5	NA
Risk (latent cancer fatalities per year)[f]	5.7×10^{-8}	NA	0.000015	NA	0.000015	0.000015	NA
Health effects from transportation[g] (projected latent cancer fatalities)							
Incident-free (involved worker)	0.0022	NA	0.0041	NA	0.0018[h]	0.0002	NA
Accidents (offsite population)[i]	0.0000021	NA	0.0000021	NA	0.0000021	0.0000021	NA
Air resources							
Nonradiological - Nitrogen oxide incremental concentration at SRS boundary (highest annual, micrograms per cubic meter)	0.033	NA	0.28	NA	0.23	0.28	NA
Water resources							
Lead (micrograms per liter) discharged to SRS streams	6.1	NA	6.0	NA	6.1	5.8	NA
Utilities (10-year totals)							
Electricity usage (megawatt-hour)	140,000	NA	181,000	NA	110,000	100,000	NA
Waste management (10-year totals)							
High-level liquid waste generation (million liters)	1.7	NA	6.9	NA	2.6	2.7	NA
Equivalent DWPF canisters	30	NA	140	NA	50	54	NA
Saltstone generation (cubic meters)	3,600	NA	18,000	NA	6,100	6,900	NA
Transuranic waste generation (cubic meters)	0	NA	0	NA	0	0	NA
Hazardous/mixed waste generation (cubic meters)	0	NA	0	NA	0	0	NA
Low-level radioactive waste generation (cubic meters)	7,700	NA	9,200	NA	6,300	5,100	NA

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a. Impact data for each alternative include management of both solution and targets.
b. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.
c. The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.
d. NA = Not applicable.
e. Assumes highly unlikely occurrence of maximum consequence accident.
f. Highest point estimate of risk during any phase of the alternative. The risk is calculated as the potential consequence of an accident (in latent cancer fatalities) multiplied by the probability per year of the accident occurring.
g. Includes transportation of associated radioactive waste.
h. Waste transportation only.
i. Maximum reasonably foreseeable latent cancer fatalities from medium probability accident based on the shipment of waste.

TE

Table 2-5. Comparison of the potential environmental impacts of the alternatives for neptunium.[a,b]

TE

Factors	Alternatives: Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[c]	Vitrification (F-Canyon)	Improving Storage
Health effects of Normal Operations							
Radiological health effects (10-year totals):							
Offsite population latent cancer fatalities	0.00027	NA[d]	0.028	NA	0.0047	0.00023	NA
Worker latent cancer fatalities	0.006	NA	0.052	NA	0.0056	0.020	NA
Health effects to offsite population from facility accidents[e] (projected latent cancer fatalities)	4.1	NA	4.1	NA	38	6.5	NA
Risk (latent cancer fatalities per year)[f]	0.0000036	NA	0.0000036	NA	0.0000036	0.000015	NA
Health effects from transportation[g] (projected latent cancer fatalities)							
Incident-free (involved worker)	0.0017[h]	NA	0.0028	NA	0.0014[h]	0.0013	NA
Accidents (offsite population)[i]	0.0000021	NA	2.0	NA	0.0000021	0.0000021	NA
Air resources							
Nonradiological - Nitrogen oxide incremental concentration at SRS boundary (highest annual, micrograms per cubic meter)	0.019	NA	0.10	NA	0.083	0.11	NA
Water resources							
Lead (micrograms per liter) discharged to SRS streams	3.0	NA	3.0	NA	3.0	3.0	NA
Utilities (10-year totals)							
Electricity usage (megawatt-hour)	142,000	NA	149,000	NA	93,000	126,000	NA
Waste management (10-year totals)							
High-level liquid waste (million liters)	1.3	NA	4.2	NA	1.5	1.0	NA
Equivalent DWPF canisters	20	NA	37	NA	27	16	NA
Saltstone generation (cubic meters)	3,600	NA	11,000	NA	4,000	2,800	NA
Transuranic waste generation (cubic meters)	0	NA	160	NA	0	0	NA
Hazardous/mixed waste generation (cubic meters)	0	NA	200	NA	0	0	NA
Low-level radioactive waste generation (cubic meters)	5,700	NA	6,400	NA	3,800	4,600	NA

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a. Impact data for each alternative include management of both solution and targets.
b. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.
c. The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.
d. NA = Not applicable.
e. Assumes highly unlikely occurrence of maximum consequence accident.
f. Highest point estimate of risk during any phase of the alternative. The risk is calculated as the potential consequence of an accident (in latent cancer fatalities) multiplied by the probability per year of the accident occurring.
g. Includes transportation of associated radioactive waste.
h. Waste transportation only.
i. Maximum reasonably foreseeable latent cancer fatalities from medium probability accident based on the shipment of waste.

TE

Table 2-6. Comparison of the potential environmental impacts of the alternatives for H-Canyon plutonium-239 solutions.[a]

TE

	Alternatives						
Factors	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[b]	Vitrification (F-Canyon)	Improving Storage
Health effects of Normal Operations							
Radiological health effects (10-year totals):							
Offsite population latent cancer fatalities	0.00025	0.00025	0.0018	NA[c]	0.041	0.00023	NA
Worker latent cancer fatalities	0.0052	0.044	0.026	NA	0.02	0.021	NA
Health effects to offsite population from facility accidents[d] (projected latent cancer fatalities)	4.1	6.5	4.1	NA	38	6.5	NA
Risk (latent cancer fatalities per year)[e]	0.0000036	0.000015	0.0000036	NA	0.0000036	0.000015	NA
Health effects from transportation[f] (projected latent cancer fatalities)							
Incident-free (involved worker)	0.0017[g]	0.0022[g]	0.0017	NA	0.0036[g]	0.0014[g]	NA
Accidents (offsite population)[h]	0.0000021	2.0	2.0	NA	0.0000021	0.0000021	NA
Air resources							
Nonradiological - Nitrogen oxide incremental concentration at SRS boundary (highest annual, micrograms per cubic meter)	0.012	0.14	0.033	NA	0.083	0.11	NA
Water resources							
Lead (micrograms per liter) discharged to SRS streams	3.2	3.3	2.9	NA	3.0	3.2	NA
Utilities (10-year totals)							
Electricity usage (megawatt-hour)	133,000	135,000	89,000	NA	151,000	124,000	NA
Waste management (10-year totals)							
High-level liquid waste (million liters)	1.2	1.3	0.55	NA	6.8	1.0	NA
Equivalent DWPF canisters	20	24	9	NA	190	17	NA
Saltstone generation (cubic meters)	3,300	3,500	1,500	NA	19,000	2,700	NA
Transuranic waste generation (cubic meters)	0	32	56	NA	0	0	NA
Hazardous/mixed waste generation (cubic meters)	0	0	63	NA	0	0	NA
Low-level radioactive waste generation (cubic meters)	5,600	7,500	6,300	NA	6,400	4,800	NA

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a. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.

b. The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.

c. NA = Not applicable.

d. Assumes highly unlikely occurrence of maximum consequence accident.

e. Highest point estimate of risk during any phase of the alternative. The risk is calculated as the potential consequence of an accident (in latent cancer fatalities) multiplied by the probability per year of the accident occurring.

f. Includes transportation of associated radioactive waste.

g. Waste transportation only.

h. Maximum reasonably foreseeable latent cancer fatalities from medium probability accident based on the shipment of waste.

TE

Table 2-7. Comparison of the potential environmental impacts of the alternatives for H-Canyon enriched uranium solutions.[a]

TE

Factors	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[b]	Vitrification (F-Canyon)	Improving Storage
Health effects of Normal Operations							
Radiological health effects (10-year totals):							
Offsite population latent cancer fatalities	0.00038	NA[c]	0.0034	0.009	0.0003	NA	NA
Worker latent cancer fatalities	0.0092	NA	0.028	0.0072	0.0072	NA	NA
Health effects to offsite population from facility accidents[d] (projected latent cancer fatalities)	0.14	NA	0.14	0.14	38	NA	NA
Risk (latent cancer fatalities per year)[e]	9.6×10^{-7}	NA	9.6×10^{-7}	9.6×10^{-7}	0.0000036	NA	NA
Health effects from transportation[f] (projected latent cancer fatalities)							
Incident-free (involved worker)	0.002[g]	NA	0.0005[g]	0.00089	0.0016[g]	NA	NA
Accidents (offsite population)[h]	0.0000021	NA	0.0000021	0.0000021	0.0000021	NA	NA
Air resources							
Nonradiological - Nitrogen oxide incremental concentration at SRS boundary (highest annual, micrograms per cubic meter)	0.053	NA	0.083	0.083	0.053	NA	NA
Water resources							
Lead (micrograms per liter) discharged to SRS streams	3.0	NA	3.0	3.0	3.0	NA	NA
Utilities (10-year totals)							
Electricity usage (megawatt-hour)	180,000	NA	40,000	42,000	140,000	NA	NA
Waste management (10-year totals)							
High-level liquid waste (million liters)	1.8	NA	0.72	1.7	1.4	NA	NA
Equivalent DWPF canisters	30	NA	7	17	130	NA	NA
Saltstone generation (cubic meters)	5,000	NA	2,000	4,800	3,900	NA	NA
Transuranic waste generation (cubic meters)	0	NA	0	0	0	NA	NA
Hazardous/mixed waste generation (cubic meters)	0	NA	0	0	0	NA	NA
Low-level radioactive waste generation (cubic meters)	6,300	NA	1,200	1,600	4,800	NA	NA

TC

TC

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a. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.
b. The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.
c. NA = Not applicable.
d. Assumes highly unlikely occurrence of maximum consequence accident.
e. Highest point estimate of risk during any phase of the alternative. The risk is calculated as the potential consequence of an accident (in latent cancer fatalities) multiplied by the probability per year of the accident occurring.
f. Includes transportation of associated radioactive waste.
g. Waste transportation only.
h. Maximum reasonably foreseeable latent cancer fatalities from medium probability accident based on the shipment of waste.

TE

Table 2-8. Comparison of the potential environmental impacts of the alternatives for plutonium and uranium stored in vaults.[a]

TE

	Alternatives						
Factors	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[b]	Vitrification (F-Canyon)	Improving Storage
Health effects of Normal Operations							
Radiological health effects (10-year totals):							
Offsite population latent cancer fatalities	0.00011	0.07	0.07	NA[c]	0.07	0.07	0.00024
Worker latent cancer fatalities	0.056	0.18	0.18	NA	0.11	0.18	0.16
Health effects to offsite population from facility accidents[d] (projected latent cancer fatalities)	0.31	4.1	4.5	NA	38	4.1	0.62
Risk (latent cancer fatalities per year)[e]	1×10^{-8}	0.0000036	0.0000022	NA	8.9×10^{-7}	0.0000036	2×10^{-8}
Health effects from transportation[f] (projected latent cancer fatalities)							
Incident-free (involved worker)	0.0052	0.0091	0.0091	NA	0.0077	0.0091	0.0062
Accidents (offsite population)[g]	2.0	2.0	2.0	NA	2.0	2.0	2.0
Air resources							
Nonradiological - Nitrogen oxide incremental concentration at SRS boundary (highest annual, micrograms per cubic meter)	0.0095	0.14	0.14	NA	0.13	0.14	0.031
Water resources							
Lead (micrograms per liter) discharged to SRS streams	3.1	3.1	3.1	NA	3.1	3.1	4.3
Utilities (10-year totals)							
Electricity usage (megawatt-hour)	147,000	190,000	190,000	NA	210,000	190,000	77,000
Waste management (10-year totals)							
High-level liquid waste (million liters)	0	8.2	8.2	NA	8.2	8.2	0
Equivalent DWPF canisters	0	61	61	NA	2,400	61	0
Saltstone generation (cubic meters)	0	22,000	22,000	NA	22,000	22,000	0
Transuranic waste generation (cubic meters)	810	1,300	1,300	NA	900	1,300	1,000
Hazardous/mixed waste generation (cubic meters)	970	1,400	1,400	NA	1,100	1,400	960
Low-level radioactive waste generation (cubic meters)	19,000	24,000	24,000	NA	19,000	24,000	23,000

TC

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a. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts. In addition, DOE calculated impacts assuming that all plutonium and uranium stored in vaults are using the listed alternative to bound the impact estimate.

b. The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.

c. NA = Not applicable.

d. Assumes highly unlikely occurrence of maximum consequence accident.

e. Highest point estimate of risk during any phase of the alternative. The risk is calculated as the potential consequence of an accident (in latent cancer fatalities) multiplied by the probability per year of the accident occurring.

f. Includes transportation of associated radioactive waste.

g. Maximum reasonably foreseeable latent cancer fatalities from medium probability accident based on the shipment of waste.

TE

Table 2-9. Comparison of the potential environmental impacts of the alternatives for Mark-31 targets.[a]

TE

Factors	Alternatives							
							Improving Storage	
	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[b]	Vitrification (F-Canyon)	Traditional construction schedule	Accelerated construction schedule
Health effects of Normal Operations								
Radiological health effects (10-year totals):								
Offsite population latent cancer fatalities	0.00006	0.00025	0.00023	NA[c]	0.00043	0.00032	0.00006	0.000032
Worker latent cancer fatalities	0.0056	0.084	0.072	NA	0.044	0.1	0.0056	0.0076
Health effects to offsite population from facility accidents[d] (projected latent cancer fatalities)	0.0089	6.5	6.5	NA	38	6.5	0.0089	0.0089
Risk (latent cancer fatalities per year)[e]	4.9×10^{-9}	0.000015	0.000015	NA	0.000015	0.000015	4.9×10^{-9}	4.9×10^{-9}
Health effects from transportation[f] (projected latent cancer fatalities)								
Incident-free (involved worker)	0.0073[g]	0.0049	0.0048	NA	0.0063	0.0053	0.0073	0.0053
Accidents (offsite population)[h]	0.0000021	2.0	2.0	NA	0.0000021	2.0	0.0000021	0.0000021
Air resources								
Nonradiological - Nitrogen oxide incremental concentration at SRS boundary (highest annual, micrograms per cubic meter)	0	0.28	0.28	NA	0.23	0.34	0	0
Water resources								
Lead (micrograms per liter) discharged to SRS streams	0	3.4	3.5	NA	6.1	3.9	0	0
Utilities (10-year totals)								
Electricity usage (megawatt-hour)	14	64,000	51,000	NA	44,000	71,000	14	1,400
Waste management (10-year totals)								
High-level liquid waste generation (million liters)	1.2	2.1	1.9	NA	3.7	2.6	1.2	0.87
Equivalent DWPF canisters	28	43	41	NA	170	53	28	18
Saltstone generation (cubic meters)	3,200	5,700	5,200	NA	10,000	7,000	3,200	2,300
Transuranic waste generation (cubic meters)	0	77	62	NA	0	93	0	0
Hazardous/mixed waste generation (cubic meters)	50	16	20	NA	34	16	50	35
Low-level radioactive waste generation (cubic meters)	29,000	18,000	18,000	NA	22,000	19,000	29,000	21,000

TC

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a. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.

b. The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.

c. NA = Not applicable.

d. Assumes highly unlikely occurrence of maximum consequence accident.

e. Highest point estimate of risk during any phase of the alternative. The risk is calculated as the potential consequence of an accident (in latent cancer fatalities) multiplied by the probability per year of the accident occurring.

f. Includes transportation of associated radioactive waste.

g. Waste transportation only.

h. Maximum reasonably foreseeable latent cancer fatalities from medium probability accident based on the shipment of waste.

TE

Table 2-10. Comparison of the potential environmental impacts of the alternatives for Mark-16 and -22 fuels.[a]

TE

	Alternatives							
							Improving Storage	
Factors	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[b]	Vitrification (F-Canyon)	Traditional construction schedule	Accelerated construction schedule
Health effects of Normal Operations								
Radiological health effects (10-year totals):								
Offsite population latent cancer fatalities	0.000016	NA[c]	0.034	0.041	0.0008	NA	0.000016	0.000011
Worker latent cancer fatalities	0.0028	NA	0.08	0.026	0.088	NA	0.0028	0.0068
Health effects to offsite population from facility accidents[d] (projected latent cancer fatalities)	0.0089	NA	4.1	4.1	38	NA	0.0089	0.0089
Risk (latent cancer fatalities per year)[e]	4.9×10^{-9}	NA	0.0000036	0.0000036	0.0000036	NA	4.9×10^{-9}	4.9×10^{-9}
Health effects from transportation[f] (projected latent cancer fatalities)								
Incident-free (involved worker)	0.0037[g]	NA	0.0054	0.0063	0.0097	NA	0.0038	0.0019
Accidents (offsite population)[h]	0.0000021	NA	0.0000021	0.0000021	0.0000021	NA	0.0000021	0.0000021
Air resources								
Nonradiological - Nitrogen oxide incremental concentration at SRS boundary (highest annual, micrograms per cubic meter)	0	NA	0.083	0.083	0.23	NA	0	0
Water resources								
Lead (micrograms per liter) discharged to SRS streams	0	NA	3	3	6.1	NA	0	0
Utilities (10-year totals)								
Electricity usage (megawatt-hour)	10	NA	79,000	83,000	89,000	NA	10	2,800
Waste management (10-year totals)								
High-level liquid waste generation (million liters)	0.57	NA	5.6	7.3	6.8	NA	0.57	0.37
Equivalent DWPF canisters	10	NA	49	68	1,000	NA	10	5
Saltstone generation (cubic meters)	1,600	NA	15,000	20,000	19,000	NA	1,600	800
Transuranic waste generation (cubic meters)	0	NA	0	0	0	NA	0	0
Hazardous/mixed waste generation (cubic meters)	20	NA	22	28	44	NA	20	10
Low-level radioactive waste generation (cubic meters)	15,000	NA	16,000	20,000	32,000	NA	15,000	7,700

TC

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a. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.

b. The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.

c. NA = Not applicable.

d. Assumes highly unlikely occurrence of maximum consequence accident.

e. Highest point estimate of risk during any phase of the alternative. The risk is calculated as the potential consequence of an accident (in latent cancer fatalities) multiplied by the probability per year of the accident occurring.

f. Includes transportation of associated radioactive waste.

g. Waste transportation only.

h. Maximum reasonably foreseeable latent cancer fatalities from medium probability accident based on the shipment of waste.

TE

Table 2-11. Comparison of the potential environmental impacts of the alternatives for other aluminum-clad targets.[a]

TE

Factors	Alternatives							
							Improving Storage	
	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[b]	Vitrification (F-Canyon)	Traditional construction schedule	Accelerated construction schedule
Health effects of Normal Operations								
Radiological health effects (10-year totals):								
Offsite population latent cancer fatalities	0.0000049	NA[c]	NA	NA	0.0034	NA	0.0000049	0.0000031
Worker latent cancer fatalities	0.00084	NA	NA	NA	0.0018	NA	0.00084	0.0018
Health effects to offsite population from facility accidents[d] **(projected latent cancer fatalities)**	0.0089	NA	NA	NA	38	NA	0.0089	0.0089
Risk (latent cancer fatalities per year)[e]	4.9×10^{-9}	NA	NA	NA	0.0000036	NA	4.9×10^{-9}	4.9×10^{-9}
Health effects from transportation[f] **(projected latent cancer fatalities)**								
Incident-free (involved worker)	0.00105[g]	NA	NA	NA	0.00072	NA	0.0011	0.00057
Accidents (offsite population)[h]	0.0000021	NA	NA	NA	0.0000021	NA	0.0000021	0.0000021
Air resources								
Nonradiological - Nitrogen oxide incremental concentration at SRS boundary (highest annual, micrograms per cubic meter)	0	NA	NA	NA	0.083	NA	0	0
Water resources								
Lead (micrograms per liter) discharged to SRS streams	0	NA	NA	NA	3	NA	0	0
Utilities (10-year totals)								
Electricity usage (megawatt-hour)	10	NA	NA	NA	5,900	NA	10	720
Waste management (10-year totals)								
High-level liquid waste generation (million liters)	0.14	NA	NA	NA	0.59	NA	0.14	0.09
Equivalent DWPF canisters	0	NA	NA	NA	15	NA	0	0
Saltstone generation (cubic meters)	390	NA	NA	NA	1,600	NA	390	200
Transuranic waste generation (cubic meters)	0	NA	NA	NA	0	NA	0	0
Hazardous/mixed waste generation (cubic meters)	10	NA	NA	NA	4	NA	10	5
Low-level radioactive waste generation (cubic meters)	4,200	NA	NA	NA	2,300	NA	4,200	2,300

TC

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a. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.

b. The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.

c. NA = Not applicable.

d. Assumes highly unlikely occurrence of maximum consequence accident.

e. Highest point estimate of risk during any phase of the alternative. The risk is calculated as the potential consequence of an accident (in latent cancer fatalities) multiplied by the probability per year of the accident occurring.

f. Includes transportation of associated radioactive waste.

g. Waste transportation only.

h. Maximum reasonably foreseeable latent cancer fatalities from medium probability accident based on the shipment of waste.

TE

Table 2-12. Comparison of the potential environmental impacts of the alternatives for Taiwan Research Reactor fuel and Experimental Breeder Reactor-II slugs.[a,b]

	Alternatives							
							Improving Storage	
Factors	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[c]	Vitrification (F-Canyon)	Traditional construction schedule	Accelerated construction schedule
Health effects of Normal Operations								
Radiological health effects (10-year totals):								
Offsite population latent cancer fatalities	0.000005	0.0002	0.0002	NA[d]	0.00017	0.00027	0.000005	0.000005
Worker latent cancer fatalities	0.0056	0.072	0.072	NA	0.021	0.088	0.0056	0.0084
Health effects to offsite population from facility accidents[e] (projected latent cancer fatalities)	(f)	6.5	6.5	NA	38	6.5	0.0089	0.0089
Risk (latent cancer fatalities per year)[g]	6.5×10^{-9}	0.000015	0.000015	NA	0.000015	0.000015	6.5×10^{-9}	6.5×10^{-9}
Health effects from transportation[h] (projected latent cancer fatalities)								
Incident-free (involved worker)	0.00033[i]	0.0034	0.0028	NA	0.00080	0.0037	0.00032	0.00032
Accidents (offsite population)[j]	0.0000021	2.0	2.0	NA	0.0000021	2.0	0.0000021	0.0000021
Air resources								
Nonradiological - Nitrogen oxide incremental concentration at SRS boundary (highest annual, micrograms per cubic meter)	0	0.28	0.28	NA	0.23	0.34	0	0
Water resources								
Lead (micrograms per liter) discharged to SRS streams	2.7	3.0	3.1	NA	3.3	3.4	2.7	2.8
Utilities (10-year totals)								
Electricity usage (megawatt-hour)	11,000	62,000	55,000	NA	27,000	69,000	11,000	8,600
Waste management (10-year totals)								
High-level liquid waste generation (million liters)	0.4	1.6	1.7	NA	1.5	2.1	0.4	0.3
Equivalent DWPF canisters	0	32	32	NA	120	41	0	0
Saltstone generation (cubic meters)	110	4,200	4,300	NA	3,300	5,500	110	77
Transuranic waste generation (cubic meters)	0	67	62	NA	0	82	0	0
Hazardous/mixed waste generation (cubic meters)	0	7	4	NA	0	6	0	0
Low-level radioactive waste generation (cubic meters)	1,300	12,000	9,600	NA	1,800	13,000	1,300	1,200

TC

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a. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.

b. DOE derived the impact data for the TRR targets and the EBR-II slugs assuming it would have to stabilize all the material; this bounds the impacts in the event of additional TRR and EBR-II material failures.

c. The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.

d. NA = Not applicable.

e. Assumes highly unlikely occurrence of maximum consequence accident.

f. Data not available.

g. Highest point estimate of risk during any phase of the alternative. The risk is calculated as the potential consequence of an accident (in latent cancer fatalities) multiplied by the probability per year of the accident occurring.

h. Includes transportation of associated radioactive waste.

i. Waste transportation only.

j. Maximum reasonably foreseeable latent cancer fatalities from medium probability accident based on the shipment of waste.

the offsite population should the worst-consequence accident occur. Alternatives involving processing operations in the chemical separations facilities and the Defense Waste Processing Facility would have higher potential consequences (in the form of additional latent cancer fatalities in the offsite population) from accidents than alternatives involving no action or improving storage, because processing operations in the chemical separations facilities and DWPF could experience accidents with higher potential consequences than facilities used simply to store radioactive material (i.e., vaults or basins).

2.4.2 AIR AND WATER RESOURCES

Chemical releases to air and water resources from normal operations associated with the alternatives would be below regulatory limits (see Sections 4.1.2 and 4.1.3, respectively). Alternatives involving processing operations in the separations facilities would have higher chemical and radiological releases than alternatives involving no action or improving storage because of the types of activities performed in the facilities (chemicals would be used to dissolve and separate the nuclear materials from fission products, corrosion products, aluminum cladding, etc.). Alternatives involving processing operations would have higher radioactive releases to air and water resources for the same reason.

L11-1

2.4.3 UTILITIES

Utilities consumption would vary among the alternatives, depending on the facility used to store or process the material. Alternatives involving storage or processing operations in the separations facilities would have significantly higher utilities usage than alternatives involving storing or repackaging materials in vaults or basins. Utilities usage would also be highly dependent on how quickly stabilization or conversion operations could occur for alternatives that would require the use of the large chemical separations facilities. These trends would apply for water, electricity, steam, and fuel usage. None of the utility usages would exceed historic rates or the current capacities of the SRS infrastructure.

2.4.4 WASTE

Estimates of the amount of high-level liquid waste generated vary greatly among the alternatives. This is true even for alternatives that would involve processing operations in the chemical separations facilities that would generate the largest amounts of high-level liquid waste. For example, the Processing to Oxide Alternative for plutonium-242 solutions would generate an estimated

120,000 liters (31,700 gallons) of high-level liquid waste as compared to 8,200,000 liters (2,170,000 gallons) from the Vitrification (F-Canyon) Alternative for plutonium and uranium vault materials. The alternatives that would involve processing operations in the chemical separations facilities would generate larger volumes of high-level liquid waste than alternatives that would involve storing or repackaging materials in vaults or basins. Volumes of high-level liquid waste from alternatives that would require chemical separations would be a function of: (1) the concentration of fissile material or special actinides contained in materials to be processed and the amounts transferred to the high-level waste tanks, and (2) the concentrations of impurities such as fission products, corrosion products, and dissolved metals (e.g., aluminum) contained and the amounts transferred to the high-level waste tanks. The higher the concentration of the constituents to be transferred to the high-level waste tanks, the greater the volume of liquid waste. In some cases, these are competing factors. For example, processing alternatives for the Mark-31 targets are likely to generate less high-level liquid waste than similar processing alternatives for the Mark-16 and Mark-22 fuels. Although the Mark-31 targets represent a larger volume of material, the Mark-16 and Mark-22 fuels contain much higher amounts of fissile material.

L11-1 At the SRS, high-level waste would be vitrified in the Defense Waste Processing Facility for eventual placement in a geologic repository. The amount of vitrified waste from each alternative is listed in Tables 2-2 through 2-12 as "equivalent DWPF canisters." A DWPF canister is a stainless-steel cylinder about 3 meters (10 feet) long and 0.6 meter (2 feet) in diameter, designed to hold vitrified high-level waste. The equivalent DWPF canister number is the estimated quantity of such canisters that would be generated after vitrifying the high-level waste produced by implementing an alternative.

Estimated volumes of other types of radioactive waste (transuranic, hazardous and mixed, and low-level) are similar among the alternatives. However, unlike the liquid high-level waste, the generation of these types of waste would depend on how long the material was maintained in its current storage condition and how quickly an alternative could be completed. This would be true both for alternatives that would involve processing operations in the chemical separations facilities and alternatives that would involve storing or repackaging materials in vaults or basins. The sooner an alternative could be completed, the smaller the amount of waste that would be generated. This would be true for the Processing, Continuing Storage, or Improving Storage Alternative. None of the estimated waste impacts would exceed historic rates or the capacity of the existing and proposed SRS waste management facilities (DOE 1995c).

2.4.5 CONTRIBUTORS TO ENVIRONMENTAL IMPACTS

In summary, the following considerations would affect the estimated environmental impacts for each alternative:

- **Processing Alternatives versus Nonprocessing Alternatives:** Alternatives that would involve operation of separations facilities [Processing to Metal, Processing to Oxide, Blending Down to Low Enriched Uranium, Vitrification (F-Canyon), and Processing and Storage for Vitrification in the Defense Waste Processing Facility] generally would result in greater impacts during the time that the separations facilities were operating. The overall effect of processing activities on 10-year impacts would depend in large part on when the processing activities began because, as indicated in Appendix D, post-stabilization storage impacts generally would be less than existing storage impacts; therefore, earlier completion of processing activities could result in impacts that were similar to or less than those from the No-Action Alternative because of the relatively low impacts of post-stabilization storage. For example, Table 4-5 indicates that the No-Action and Vitrification (F-Canyon) Alternatives for H-Canyon plutonium-239 solutions would yield similar estimates of the atmospheric population dose. Table D-21 shows that the annual atmospheric population dose after stabilization (post-stabilization) storage would be less than that from the existing storage configuration and that the annual atmospheric population dose during the stabilization activities would be greater than that from the existing storage configuration. If vitrification (which would take 6 months) were completed early in the 10-year period, the temporary increase in atmospheric population dose due to the conversion activities would be offset during the remainder of the period by the reduced post-stabilization storage dose. In this case, therefore, an alternative that would involve separations activities would result in an estimate for one environmental factor that is similar to that of No Action for 10 years. TC

- **Choice of Separations Facilities:** Although the F-Area and H-Area Separations Facilities are similar, their operating parameters are not identical. Historic information was used to estimate impacts resulting from operations of these facilities, and environmental factors would vary depending on the particular facility used. For example, Table 4-5 indicates that processing the H-Canyon plutonium-239 solutions to metal in F-Area could result in a lower dose to the maximally exposed individual than processing the same solutions to oxide in H-Area. Table 2-13 compares the separations facilities in F- and H-Areas to environmental factors.

Table 2-13. Comparison of environmental factors between separations facilities in F- and H-Areas under operating conditions.[a,b]

TC

Factor	Greater for F-Area, H-Area, or Similar[c]	Factor	Greater for F-Area, H-Area, or Similar
Atmospheric MEI[d]	H-Area	Onsite CO concentration - 8 hr. average	H-Area
Liquid MEI dose	H-Area	Onsite NO_X concentration - 1 hr. average	F-Area
Total MEI dose	H-Area	Onsite SO_2 concentration - 8 hr. average	H-Area
Atmospheric population dose	H-Area	Onsite HNO_3 concentration - 8 hr. average	F-Area
Liquid population dose	H-Area	Onsite CO_2 concentration - 8 hr. average	H-Area
Total population dose	H-Area	Average number of radiation workers	Similar[c]
Offsite CO[e] concentration - 1-hr. average	H-Area	Collective worker dose	F-Area
Offsite CO concentration - 8-hr. average	H-Area	Water usage	F-Area
Offsite NO_X[f] concentration - annual average	F-Area	Electricity usage	Similar[c]
Offsite SO_2[g] concentration - 3-hr. average	H-Area	Steam usage	Similar[c]
Offsite SO_2 concentration - 24-hr. average	H-Area	Fuel usage	Similar[c]
Offsite SO_2 concentration - annual average	H-Area	High-level liquid waste generation	Similar[c]
Offsite gaseous fluorides - 12-hr. average	F-Area	Equivalent DWPF[i] canisters	F-Area
Offsite gaseous fluorides - 24-hr. average	F-Area	Saltstone generation	Similar[c]
Offsite gaseous fluorides - 1-week average	F-Area	TRU[j] waste generation	Similar[c]
Offsite gaseous fluorides - 1-month average	F-Area	Hazardous/mixed waste generation	H-Area
Offsite HNO_3[h] concentration - 24-hr. average	F-Area	Low-level waste generation	F-Area
Offsite HNO_3 concentration - annual average	F-Area		

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a. Source: WSRC (1995a).
b. Assumes 12 months operation of both the applicable Canyon and B-Line.
c. In this table, "similar" means that the difference between the values for F- and H-Areas is less than 50 percent.
d. MEI = maximally exposed individual.
e. CO = carbon monoxide.
f. NO_X = nitrogen oxides.
g. SO_2 = sulfur dioxide.
h. HNO_3 = nitric acid.
i. DWPF = Defense Waste Processing Facility.
j. TRU = transuranic.

- **Estimated Durations for Activities within an Alternative:** The estimated durations for active phases would have a fundamental effect on the estimated 10-year impacts. More time required to achieve stabilization would result in an increase in impacts while the stabilization occurred; however, if the stabilization was completed earlier in the 10-year period, the relatively low impacts from post-stabilization storage would help offset the increased impacts resulting from stabilization, as discussed above. Therefore, activities that would require relatively long periods of time would increase the impacts and could extend the total stabilization time such that all impacts would not occur within the 10-year time period. For example, Tables 4-5 and 4-8 show that the estimated impacts from vitrifying the H-Canyon plutonium-239 solutions would be less than those from vitrifying the Mark-31 targets, even though both contain plutonium-239 material and would be vitrified in the same manner. The Mark-31 targets, however, would require a longer stabilization time (because the targets would have to be dissolved) and thus could have greater impacts over the 10-year period.

TC

The effects of these "drivers" on environmental impacts would depend on the environmental factor being considered (e.g., health effects, waste generation, air quality, water quality, utilities). In some cases, one or more of the drivers could offset the effects of other drivers (i.e., one driver could serve to increase impacts while another driver decreased impacts, so the net result would be unchanged). This could result in several alternatives for a given material having the same estimated impacts even though different stabilization activities were occurring. Chapter 4 describes the variations in impacts among alternatives.

2.5 Other Factors

The selection of alternatives for the stabilization of SRS nuclear materials depends in part on existing technology (or on technology that DOE could develop quickly), the capabilities of existing SRS facilities, and the extent to which the actions would support long-term storage objectives. Consistent with a comprehensive review of options for plutonium disposition, DOE will consider the technical, nonproliferation, environmental, budgetary, and economic aspects of each alternative in each scenario before it selects any alternative for implementation.

TC

In addition to comparing alternatives against the environmental criteria listed in Section 2.4, DOE has compared other factors related to the stabilization of nuclear materials. These factors are representative of issues addressed by the National Academy of Sciences in its study of the management and disposition of plutonium (NAS 1994), the Office of Technology Assessment plutonium study (OTA 1993), and comments received during the EIS scoping period.

2.5.1 NEW FACILITIES REQUIRED

DOE would have to build one or more of the following new facilities to implement some alternatives:

- F-Canyon Vitrification Facility - This facility would be required for the Vitrification (F-Canyon) Alternative. Its use is the preferred alternative for the americium and curium solution in F-Canyon. DOE would build this facility by modifying the space in F-Canyon previously called the Multi-Purpose Processing Facility.

- Dry Storage Facility - This facility would be required for the Improving Storage Alternative for fuel and targets currently stored in reactor disassembly and canyon basins. For the purposes of determining environmental impacts, DOE assumed it would construct this new facility on an undeveloped area of the SRS (see Section 4.3 and Appendix C); if practical, however, DOE would seek to minimize environmental impacts by siting the facility on a previously developed area of the SRS.

- Uranium Solidification Facility (USF) - This facility would be required for the Processing to Oxide Alternative for highly enriched uranium solutions in H-Area and the Mark-16 and -22 fuel stored in basins. The USF is partially constructed and would be completed in H-Canyon, occupying that part that previously contained the "old HB-Line." However, the use of this facility is not the preferred alternative for any of these materials. TC

- Actinide Packaging and Storage Facility - The facility would be required for the Processing Oxide and Processing to Metal Alternatives for plutonium (vault materials, plutonium-239 solutions in H-Canyon, and Mark-31 targets in basins) and for the Improving Storage Alternative for plutonium and uranium stored in vaults. It would also be required for the Processing to Oxide Alternative for neptunium-237 materials and the Vitrification (F-Canyon) Alternative for the americium and curium solutions, which are the preferred alternatives for those materials (see Table 2-1). DOE's preferred method would be to construct a new facility in F-Area near Building 247-F.

- A storage building for low enriched uranium oxide would be required for the Blending Down to Low Enriched Uranium Alternative for highly enriched uranium solutions and the Mark-16 and -22 fuels. The new building would be essentially a warehouse in F-Area. Its size and storage capacity would depend on the amount of highly enriched uranium blended down to low enriched uranium oxide. Blending Down to Low Enriched Uranium is the preferred alternative for highly enriched uranium solutions.

Appendix C contains more detailed descriptions of these proposed facilities, including the F-Canyon Vitrification Facility and the Uranium Solidification Facility even though they would occupy space in existing facilities. These facilities would require extensive modifications of existing buildings, would use little or none of the equipment presently in those spaces, and would represent a complete change in the use of that part of the facility. TC

In addition to these new facilities, some of the alternatives would require minor modifications to existing facilities. Examples include modifying FB-Line to provide a plutonium oxide conversion capability or a bagless transfer and packaging capability, modifying F- or H-Canyon to provide the capability to transfer solutions to containers or packages for loading and unloading, or modifying storage racks in basins to consolidate material. Because such modifications would not be extensive and would not change the primary functions of the facilities, this EIS does not describe those facilities as new. The description of each alternative in this chapter includes any modifications that would be required to existing facilities.

2.5.2 SECURITY AND NONPROLIFERATION

Every U.S. administration since 1945 has recognized that preventing the spread of nuclear weapons must be a fundamental national security and foreign policy objective of the United States. The current U.S. nuclear weapons nonproliferation policy is summarized in the *White House Fact Sheet on Nonproliferation and Export Control Policy*, dated September 27, 1993. This policy makes it clear that the United States does not encourage the civil use of plutonium and, accordingly, does not itself engage in plutonium reprocessing for either nuclear power or nuclear explosives. In addition, the United States will seek to eliminate where possible the accumulation of stockpiles of highly enriched uranium and plutonium.

L11-11

The range of management alternatives evaluated for nuclear materials that contain plutonium and highly enriched uranium varies. For example, alternatives for radioactive solutions and irradiated fuels that contain highly enriched uranium include processing the materials to a highly enriched oxide, blending them with depleted uranium to produce a low enriched uranium oxide, transferring highly enriched uranium solutions to the high-level waste tanks for vitrification in the Defense Waste Processing Facility, packaging and placing the fuel in dry storage, and continuing storage of the materials in their current forms. Of these alternatives, only two, Processing and Storage for

Vitrification in the Defense Waste Processing Facility and Blending Down to Low Enriched Uranium, would "eliminate where possible the accumulation of stockpiles of highly enriched uranium." Both would involve chemical processing of the fuels. Blending Down to Low Enriched Uranium is DOE's preferred alternative. All of the other alternatives, including dry storage of irradiated fuel, would maintain the highly enriched uranium in some form as part of the SRS inventory.

Materials that contain significant amounts of plutonium include metal and oxides stored in vaults, radioactive solutions in H-Canyon, and Mark-31 targets in reactor basins. Alternatives for these materials include continuing storage; chemical separation of the plutonium and processing to a metal, oxide, or glass; repackaging of the vault materials; and dry storage of the Mark-31 targets. None of the alternatives would denature or eliminate the plutonium from the current inventory (i.e., it would still exist in some form).

Vitrification in the Defense Waste Processing Facility would result in a form with high radiation levels and potentially dilute amounts of plutonium. This is the only alternative that could eliminate proliferation of this material. The process associated with the alternative would dilute the material over such a large volume of high-level radioactive waste that the plutonium-239 would be essentially unrecoverable. The other alternatives would involve various concentrations (i.e., Improving Storage, Processing to Metal, Processing to Oxide, and Vitrification in F-Canyon) of plutonium-239. More than 90 percent of the plutonium is already in either metal or oxide form. The proposed alternatives for plutonium solutions and the Mark-31 targets would contribute only a small amount to the existing inventory. L11-11

Because of the potential international sensitivities of processing and consolidating plutonium from the SRS inventory, the Secretary of Energy has committed that any separated or stabilized plutonium-239 and highly enriched uranium would be prohibited from use for nuclear explosives purposes (DOE 1994c). This is consistent with U.S. nuclear nonproliferation policy.

DOE is evaluating management alternatives for the disposition of weapons-usable fissile materials, including plutonium (see Section 1.6). These alternatives include methods to eliminate stockpiles of plutonium that might be surplus to national defense needs. DOE is also preparing a Programmatic EIS on alternatives for stewardship of the nuclear weapons stockpile (see Section 1.6). The Stockpile Stewardship and Management Programmatic EIS will include activities that could require the use of the plutonium-242 inventory at the SRS. This Interim Management of Nuclear Materials EIS is intended to support only DOE decisions on the interim management of plutonium-242 at the SRS, and not to determine whether the Department will ultimately use the material. The stockpile stewardship program is closely aligned with the objectives of the U.S. nuclear nonproliferation policy. L11-11

2.5.3 IMPLEMENTATION SCHEDULE

The sequence charts in Sections 2.2 and 2.3 show that the alternatives for many different materials would involve the use of the same facilities. DOE developed these charts to show the activities that each alternative would require, independent of one another. However, DOE anticipates that the implementation of an alternative for every type of material would use many of the same facilities to achieve stabilization. DOE would establish specific schedules for these facilities that would identify the order (or priority) in which it would stabilize the materials. In other words, the integration of the selected alternatives into a stabilization program would affect the implementation schedule. To examine these potential impacts and to illustrate the effect on an overall implementation schedule, DOE constructed four representative combinations of alternatives (called "scenarios").

- No-Action Scenario - The impacts projected for this scenario could occur if current storage practices continued over the 10-year period.

- Minimum Processing Scenario - The impacts from this scenario would be the sum of the impacts from the minimal operation of the separations facilities that achieved stabilization of materials. TC

- Preferred Scenario - The impacts from this scenario would be the sum of the impacts from the preferred alternative for each type of material (as described in Sections 2.2 and 2.3) over the 10-year period.

- Comparative Scenario - The estimated impacts from this scenario would be the highest overall for the 10-year period.

DOE uses these four scenarios in Chapter 5 to estimate the cumulative impacts from actions proposed in this EIS. Table 2-14 lists the alternatives that comprise each scenario.

Using information from the sequence charts for the alternatives, DOE developed an integrated schedule for each scenario. These schedules consider factors that the sequence charts do not include.

Table 2-14. Composition of management scenarios.

Material	No Action	Minimum Processing	Preferred	Comparative
Stable material	Continuing Storage	Continuing Storage	Continuing Storage	Continuing Storage
Plutonium-242	Continuing Storage	Processing to Oxide	Processing to Oxide	Vitrification (F-Canyon)
Americium and curium	Continuing Storage	Solutions - Vitrification (F-Canyon) Targets - Continuing Storage	Solutions - Vitrification (F-Canyon) Targets - Continuing Storage	Processing to Oxide
Neptunium	Continuing Storage	Processing to Oxide	Processing to Oxide	Processing to Oxide
H-Canyon plutonium-239 solutions	Continuing Storage	Processing to Oxide	Processing to Oxide	Processing to Metal
H-Canyon enriched uranium solutions	Continuing Storage	Blending Down to Low Enriched Uranium	Blending Down to Low Enriched Uranium	Blending Down to Low Enriched Uranium
Plutonium and uranium stored in vaults	Continuing Storage	Improving Storage	Processing to Metal[a] Improving Storage[a] Processing to Oxide[a] Vitrification (F-Canyon)[a]	Vitrification (F-Canyon)
Mark-31 targets	Continuing Storage	Improving Storage	Processing to Metal	Vitrification (F-Canyon)
TC Mark-16 and -22 fuels	Continuing Storage	Improving Storage	Continuing Storage	Processing and Storage for Vitrification (DWPF)[b]
Other aluminum-clad fuel and targets	Continuing Storage	Improving Storage	Continuing Storage	Processing and Storage for Vitrification (DWPF)
Failed TRR fuel and EBR-II slugs[c]	Continuing Storage	Improving Storage	Processing to Metal	Vitrification (F-Canyon)

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a. For the plutonium and uranium stored in vaults, there are four preferred alternatives. DOE will choose the appropriate alternative for a particular solid based on results of the material inspection, as discussed in Section 2.3.3. The analysis in this EIS presents the impacts from Processing to Metal (which would produce the greatest impacts) as a conservative estimate of impacts.

b. DWPF = Defense Waste Processing Facility.

c. TRR = Taiwan Research Reactor; EBR = Experimental Breeder Reactor.

For example, the chemical stabilization of two or more types of nuclear material using the same facility could require some time between the implementation of the alternatives to complete the following actions in preparation for the next stabilization activity:

- Flush systems of residual levels of other nuclear materials and impurities

- Prepare for any differences in facility operation (e.g., one glovebox line versus another)

- Make equipment or hardware changes to handle a different type of material (e.g., dry storage of long fuel tubes versus short target slugs)

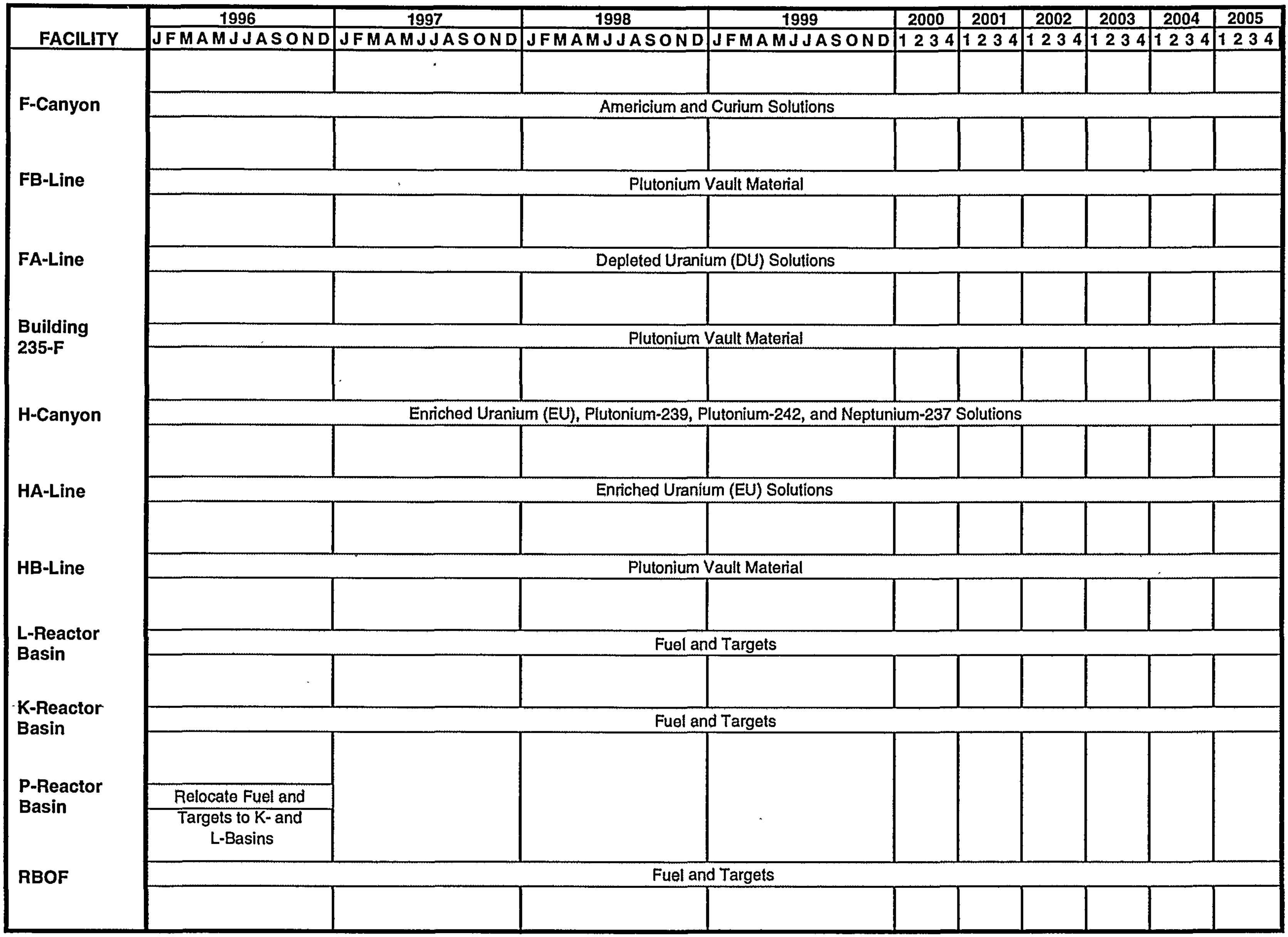
- Verify the operational readiness of a facility

The integrated schedules show the relative timing of activities expected to occur in the major SRS nuclear facilities used to store and stabilize materials. The schedules start from the time DOE would make decisions as a result of this EIS. To construct the schedules DOE used 1996 as the starting point. The schedules illustrate, in a general fashion, the required facilities and the time it could take DOE to stabilize the nuclear materials that are the subject of this EIS. The following sections discuss the assumptions used to construct each of the schedules and present points for consideration in the decision process. Section 2.6 presents estimated costs for an integrated stabilization program using each scenario.

2.5.3.1 No-Action Scenario

TC

The No-Action Scenario (see Figure 2-3) involves maintaining the storage of the nuclear materials in their current storage locations and configurations. DOE could repackage some of the materials, but would return them to their current storage environments (i.e., wet storage for fuel and targets, tank storage for solutions, and dry vault storage for metal and oxide packages). In addition, DOE could consolidate the storage of some materials in vaults, tanks, or basins to reduce operating and maintenance costs, but the physical and chemical forms and the storage environments would remain the same. The integrated schedule indicates that this scenario would result in the maintenance of nuclear materials indefinitely in almost all of the major nuclear facilities currently in use. DOE would move the material in the P-Reactor Basin to the other basins because they have the necessary storage capacity; this would reduce the costs of managing the material. The canyons and vaults do not have enough storage capacity to support the consolidation of solutions and materials to the extent that DOE could use a single canyon or a reduced number of vaults. DOE would not make any substantive effort to alleviate the vulnerabilities identified with current storage conditions.



TC

Figure 2-3. Integrated Schedule - No-Action Scenario.

2.5.3.2 Preferred Scenario

The following paragraphs discuss the actions DOE would take in the facilities that this scenario (see Figure 2-4) would affect:

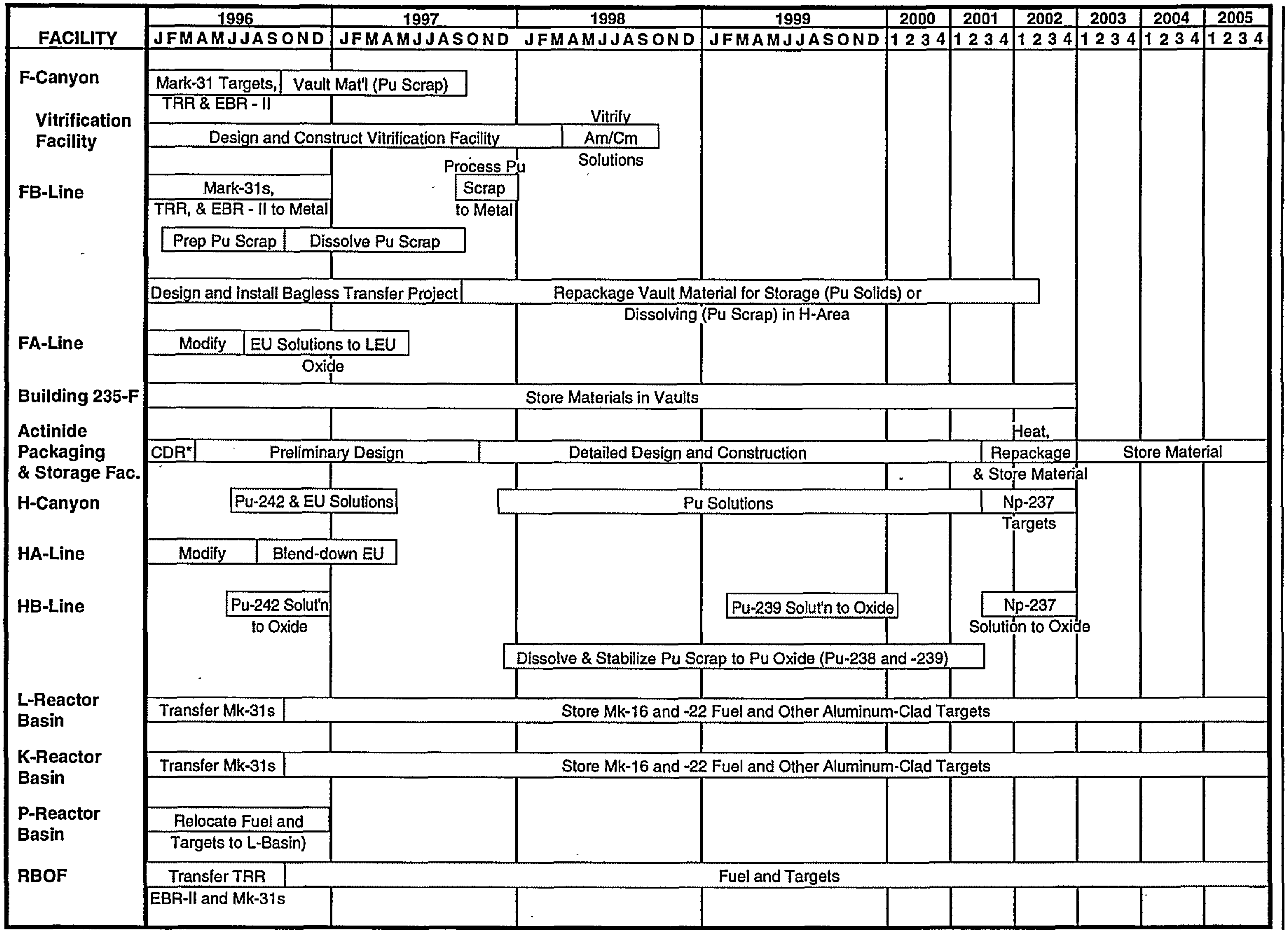
- F-Canyon and FB-Line - DOE would stabilize the Mark-31 targets, failed Taiwan Research Reactor fuel, and Experimental Breeder Reactor-II slugs using the F-Canyon and FB-Line facilities. These would be the first materials stabilized by these facilities because: (1) DOE must modify the F-Canyon to stabilize the tank of americium and curium solution, (2) removal of the materials from the basins would reduce the amount of radioactivity released to the surrounding water, and (3) the operations required for the stabilization of the materials would be identical to those for the processing of the F-Canyon plutonium solutions, which DOE is completing (see Section 1.6). Dissolving the targets and canisters would be the only additional step in comparison to the existing solutions.

  In conjunction with the stabilization of the Mark-31 targets, failed TRR fuel, and EBR-II slugs, DOE would repackage, dissolve, and chemically stabilize some of the plutonium vault materials using F-Canyon and FB-Line because both processes would require the same facility operations. DOE would use either F-Canyon or FB-Line to dissolve the vault materials (F-Canyon for materials containing sand, slag, and crucible process residues that contained potentially reactive calcium and fluorides and FB-Line for plutonium metal sweepings and turnings from machining and manufacturing operations). FB-Line would stabilize the plutonium-bearing vault materials by converting them to a solid metal form.

  TC

  DOE would stabilize the tank of americium and curium solution by vitrification as soon as it completed modifications to F-Canyon. The portions of F-Canyon that would require modification to install the vitrification equipment are not required to stabilize the Mark-31 targets or the vault materials.

- FA-Line - DOE would transport the depleted uranium solutions currently stored in F-Area to H-Area to blend them with highly enriched uranium solutions. After blending, the resulting low enriched uranium solutions would be transported back to FA-Line and converted to an oxide, which would be packaged in drums and stored in an existing warehouse. DOE could construct a new warehouse to consolidate storage of the low enriched uranium oxide. If DOE were to determine the disposition of highly enriched uranium before or during the blend-down operations, it could not convert the low enriched uranium solutions to an oxide.



*CDR = Conceptual Design Report

Figure 2-4. Integrated Schedule - Preferred Scenario.

TC

However, DOE could dispose of the low enriched uranium in either a solution or an oxide form (e.g., by commercial sale); packages and containers have been authorized for the transport of both forms. If the material remained at the SRS, DOE would convert it to an oxide in FA-Line.

- Actinide Packaging and Storage Facility - DOE would construct the Actinide Packaging and Storage Facility in F-Area to (1) support repackaging and stabilization of vault materials, (2) provide adequate storage capacity to consolidate all vault materials (thereby enabling the phaseout of the vaults in HB-Line, FB-Line, and Building 235-F), and (3) provide a surveillance and inspection capability to meet International Atomic Energy Agency requirements. DOE would move vault materials and repackaging operations to the Actinide Packaging and Storage Facility when construction was complete. When the facility was available, DOE would stabilize the neptunium solutions in H-Area (see HB-Line discussion below).

- H-Canyon - As described above, DOE would blend depleted uranium solutions from F-Area with highly enriched uranium solutions in H-Area to produce low enriched uranium solutions for stabilization in FA-Line. This would occur for several reasons: (1) it would alleviate the need for continued storage of the highly enriched uranium solutions (which have an inherent criticality concern due to the fissile material content and lack of geometry control afforded by the size of the tanks), (2) it would provide near-term reductions in the volume of uranium solutions stored in F- and H-Areas, and (3) it would require minimal operations in H-Area to support. DOE would begin blending the solutions as soon as it could make modifications to H- and F-Area facilities to enable the loading and unloading of containers from the storage tanks. TC

  DOE would dissolve the obsolete neptunium targets, store the material in the tank that contains neptunium solution, and stabilize the material in HB-Line. DOE would not dissolve the targets until it could stabilize the neptunium solution in HB-Line. This would minimize the time the neptunium would have to remain in solution.

- HB-Line - DOE would stabilize the plutonium-242 solutions stored in H-Canyon by converting them to an oxide using HB-Line; these solutions would be processed first because the required portions of HB-Line are also used to process plutonium-238 for NASA (see Chapter 1). The solutions would be stabilized after the plutonium-238 work was complete; this would minimize the time the plutonium-242 solutions would remain in tanks.

At present, DOE is not operating the portions of HB-Line required to stabilize the plutonium-239 solutions or plutonium vault materials (i.e., portions of Phase I and Phase II; see Appendix C). Due to limitations in personnel resources and vault space, DOE would not use HB-Line to stabilize these materials until it could transfer personnel from F-Area and could repackage some of the material stored in the vaults to create additional space.

DOE would not stabilize the neptunium solutions stored in H-Canyon until it completed the construction of the proposed Actinide Packaging and Storage Facility. Radiation levels associated with the neptunium oxide require heavy shielding and remote handling soon after conversion due to the in-growth of protactinium-233 (a daughter product of neptunium-237 that decays by emitting high-energy gamma rays). Historically, neptunium oxide was produced and stored at the SRS for a limited time (i.e., months) until it could be fabricated into metal reactor targets. Therefore, DOE would not stabilize the neptunium until the Actinide Packaging and Storage Facility was available to support shielded or remote packaging and storage to minimize exposure to workers.

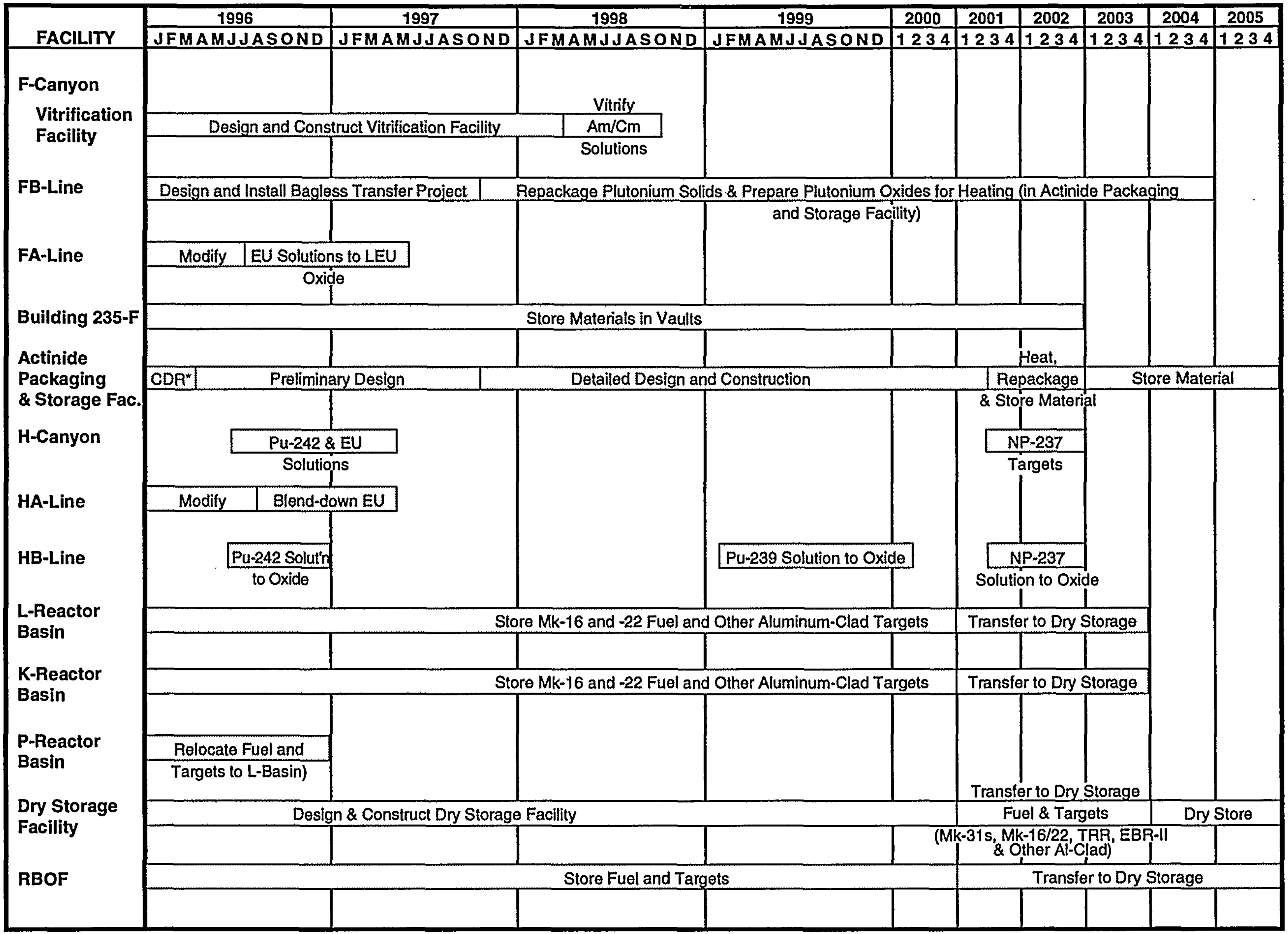
TC

- Basins - DOE would continue projects to upgrade the physical condition of the L- and K-Reactor Basins. The Department would move the material stored in the P-Reactor Basin to the L- or K-Reactor Basin. The Mark-31 targets, failed Taiwan Research Reactor fuel, and Experimental Breeder Reactor-II slugs would be loaded and transported by cask to the F-Canyon for dissolving and chemical stabilization. All other basin materials would remain in wet storage until DOE selected a stabilization alternative (see Section 2.7). DOE could place additional fuel or targets with failed cladding in canisters to limit the spread of insoluble radioactive contamination; this would depend on the stabilization alternative that DOE selected and the estimated time the materials would have to remain in wet storage.

2.5.3.3 Minimum Processing Scenario

DOE would complete actions under the Minimum Processing Scenario (see Figure 2-5) using much the same approach as that described for the Preferred Scenario. However, DOE would not use chemical separation and processing to stabilize nuclear materials other than tanks of existing solutions. The following paragraphs describe the differences from the Preferred Scenario.

- F-Canyon and FB-Line - DOE would not dissolve Mark-31 targets, failed Taiwan Research Reactor fuel, or Experimental Breeder Reactor-II slugs in F-Canyon or process them to metal in FB-Line. In addition, DOE would not dissolve or process plutonium-bearing vault



*CDR = Conceptual Design Report

Figure 2-5. Integrated Schedule - Minimum Processing Scenario.

TC

materials to remove other radioactive or hazardous constituents. The vault materials would be repackaged in FB-Line on completion of the bagless transfer project. The vault materials would be heated and repackaged, if necessary, in the Actinide Processing and Storage Facility. Vault materials would eventually be transferred to that facility as described for the Preferred Scenario. Initial repackaging work would occur in FB-Line after the installation of bagless transfer equipment. As a variation, DOE could use both FB-and HB-Lines for initial repackaging, but the Actinide Packaging and Storage Facility would have to provide the necessary storage space and heating capability to achieve a configuration that met the DOE standard for storage of plutonium metals and oxides (DOE 1994h).

- H-Canyon and HB-Line - DOE would convert the solutions stored in H-Canyon and H-Area (plutonium-242, plutonium-239, highly enriched uranium, neptunium-237) to stable oxide forms. Under this scenario, the only material that DOE would dissolve in H-Canyon would be nine obsolete neptunium reactor targets. DOE would dissolve and recover the neptunium in the targets primarily for programmatic reasons. (At present, the targets must be stored in large shielded racks due to radiation levels from protactinium-233.) The targets could be dissolved at any time before the conversion of the neptunium-237 solution to oxide. However, DOE would probably dissolve the targets immediately before converting the solution to an oxide; this would minimize the time the neptunium would be in a liquid form and the subsequent in-growth of decay products such as protactinium-233. DOE would not convert the neptunium solution to an oxide until the Actinide Processing and Storage Facility was built, as explained above. The highly enriched uranium, plutonium-239, and plutonium-242 solutions would be stabilized in the same manner, sequence, and priority described above for the Preferred Scenario.

TC

- Actinide Packaging and Storage Facility - DOE would construct this facility for the reasons and in the same period described above for the Preferred Scenario.

- Basins - As with the Preferred Scenario, DOE would move materials stored in the P-Reactor Basin to the K- and L-Reactor Basins to minimize the number of basins requiring continued management. Similarly, DOE could move fuel and target materials from the H- and F-Canyon basins to the K- or L-Reactor Basin. The projects to upgrade K- and L-Reactor Basins would be completed as described above. In addition, DOE could can material with failed cladding and return it to wet storage to minimize the spread of insoluble contamination. DOE would store materials in the basins until the construction of the proposed Dry Storage Facility.

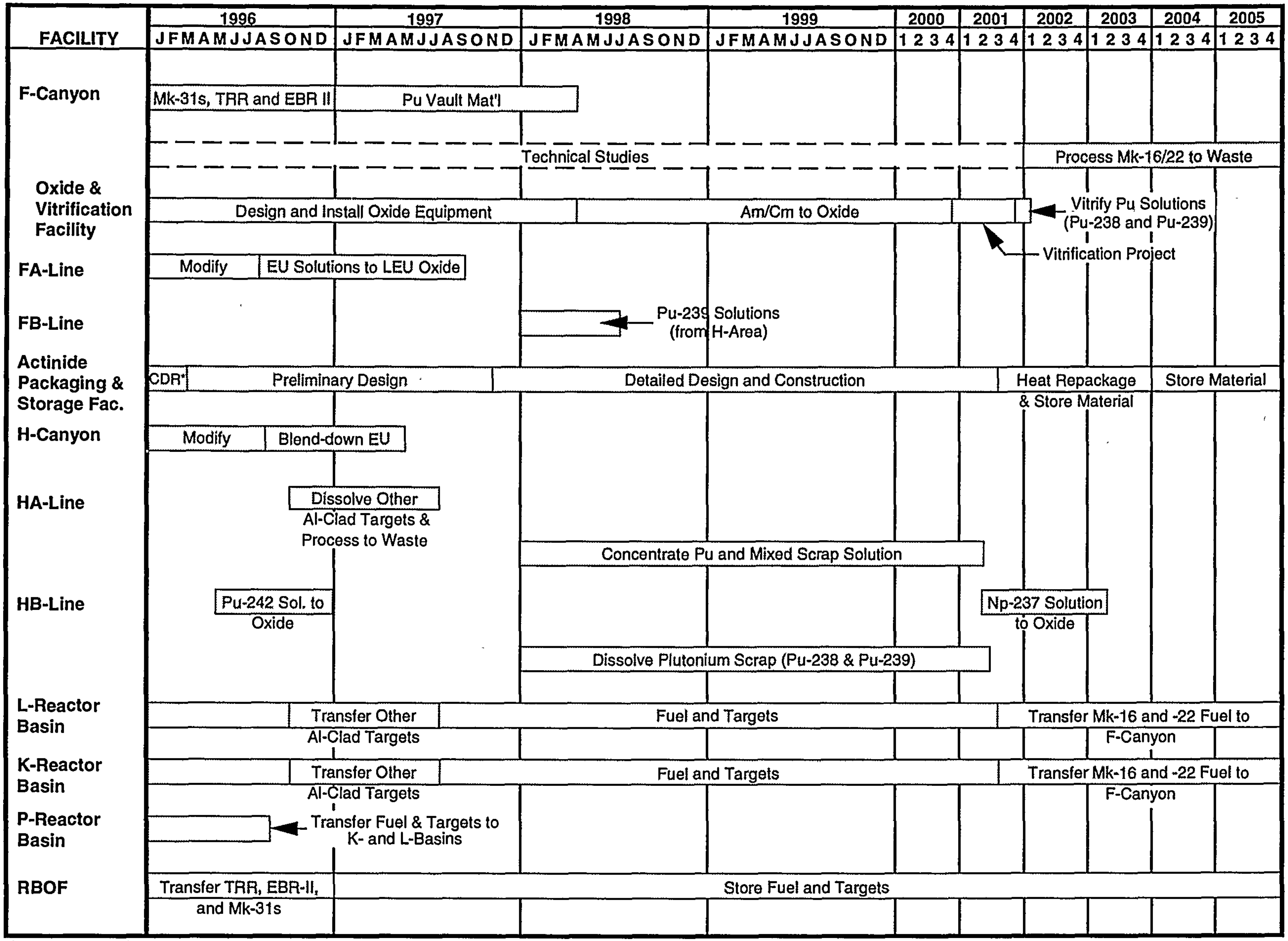
- Dry Storage Facility - DOE would design and construct this facility on an accelerated schedule, depending on Congressional authorization and funding. Construction would take approximately 5 years, at which time DOE would begin the transfer of fuel and targets from the reactor and canyon basins. DOE would need approximately 3 years to heat, can, and dry store the fuel and targets stored in the basins, and would dry store the failed fuel and canned material in the Receiving Basin for Offsite Fuel (i.e., Taiwan Research Reactor fuel, Experimental Breeder Reactor-II slugs, and Mark-31 targets). DOE could place the other material in the Receiving Basin for Offsite Fuel in dry storage, including fuel and targets categorized as Stable, depending on the outcome of other National Environmental Policy Act reviews and studies (see Section 1.6).

2.5.3.4 Comparative Scenario

This scenario (Figure 2-6) is radically different from the other scenarios for several reasons. First, in developing this scenario, DOE identified the alternative that potentially would create the greatest environmental impacts for each material category and subcategory. For this reason, it would be very unlikely for DOE to support the implementation of this combination of alternatives. However, the Comparative Scenario is bounding for many other possible combinations. In fact, it illustrates why DOE does not prefer to use alternatives that would require several years of technical studies and that would not result in material stabilization within the next 10 years. TC

- F-Canyon and FB-Line - DOE would transfer the Mark-31 targets, failed Taiwan Research Reactor fuel, and Experimental Breeder Reactor-II slugs from the basins and dissolve them in F-Canyon. DOE would chemically separate and store the plutonium-239 in the targets in solution until it could design and modify a portion of the F-Canyon to install vitrification equipment. Similarly, DOE would dissolve plutonium-bearing vault materials and store the plutonium solutions in F-Canyon tanks. When the vitrification equipment was installed, DOE would vitrify the plutonium-239 from the basin and the vault materials.

  Before vitrifying the plutonium solutions, DOE would design and modify a portion of F-Canyon to provide the capability to convert the americium and curium solution to an oxide. DOE would first convert the americium and curium to an oxide because of the high specific activity and radiological hazard posed by continued storage in a liquid form. In other words,



*CDR = Conceptual Design Report

Figure 2-6. Integrated Schedule - Comparative Scenario.

TC

DOE would place greater priority on stabilizing the americium and curium solution than the plutonium material. However, DOE would dissolve the Mark-31 targets as soon as possible to alleviate their further degradation in the reactor basin and minimize the spread of insoluble contamination in the basins. DOE would vitrify the plutonium solutions after converting the americium and curium solution because it would modify the same portion of F-Canyon and use some of the same equipment to perform both operations.

After completing facility modifications to FA- and HA-Lines to enable the loading and unloading of solutions into shipping containers, DOE would blend highly enriched uranium solutions in H-Area with depleted uranium solutions in F-Area and convert the resulting low enriched uranium solutions to an oxide using FA-Line. As described for the Preferred Scenario, DOE could make dispositions decisions on the highly enriched uranium solutions before completing this alternative. If the disposition chosen for this material required removal from the SRS, DOE could either transport it as a low enriched uranium solution or an oxide. Licensed containers are available to transport the material in either form. If the material were to be stored at SRS for an extended period, DOE would convert the solution to an oxide to alleviate the potential for liquid spills.

DOE would develop or procure a container to transport plutonium-239 solutions from H-Canyon to F-Canyon. In addition, DOE would modify each facility to enable the loading and unloading of the shipping containers. After transferring the plutonium-239 solutions to F-Canyon, DOE would use FB-Line to convert the solutions to plutonium metal, package the metal, and store it in a vault until it could construct the Actinide Packaging and Storage Facility. Unlike the Preferred and Minimum Processing Scenarios, DOE would not modify FB-Line to provide a bagless transfer capability, but would package the small amount of metal using existing capabilities and wait for the availability of the Actinide Storage and Processing Facility to achieve a configuration in compliance with the storage standard. TC

- Actinide Packaging and Storage Facility - As described for the Preferred and Minimum Processing Scenarios, DOE would design and construct this facility and use it to store the plutonium metal, plutonium glass, americium and curium oxide, and neptunium oxide that result from stabilization actions.

- H-Canyon and HB-Line - As described for the Preferred and Minimum Processing Scenarios, DOE would blend the highly enriched uranium solutions in H-Area with the depleted uranium solutions in F-Area to produce solutions containing low enriched uranium, which it would convert to an oxide or dispose of.

DOE would convert the plutonium-242 solutions to an oxide using HB-Line as soon as it completed processing the plutonium-238. The plutonium-242 oxide would be packaged and transported to F-Area, where the oxide would be redissolved and vitrified in the modified portion of F-Canyon.

TC DOE would dissolve plutonium vault materials in H-Canyon or HB-Line, depending on the nonplutonium constituents and the volume of material to be dissolved. The resulting solutions would be transported to F-Canyon and vitrified, as described above.

When the Actinide Packaging and Storage Facility was complete, DOE would: dissolve the nine obsolete reactor targets containing neptunium-237 in H-Canyon, recover the neptunium-237 and add it to the neptunium-237 in solution; operate HB-Line to convert the solution to an oxide; package the oxide and transport it to the Actinide Packaging and Storage Facility; and heat or repackage the oxide as necessary to meet storage requirements.

2.5.4 TECHNOLOGY AVAILABILITY AND TECHNICAL FEASIBILITY

This factor relates to the extent that technology development would be required and its likelihood of success. Processing to Metal in F-Area and Processing to Oxide in H-Area represent the most technically proven of the stabilization alternatives; they would use existing technology and equipment. TE The Vitrification (F-Canyon) and Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternatives appear to be technically feasible, but would require increasing amounts of technology development. Dry storage would involve the most technology development.

In general, the technical uncertainty would increase as the stabilized form differed from that historically produced. There would also be technical uncertainty about the continued storage of the TE solutions under the Continuing Storage Alternative as a result of radiation and chemically induced changes in the solution chemistry and form.

2.5.5 LABOR AVAILABILITY AND CORE COMPETENCY

TE There would be differences between the level of personnel knowledge and training required for each alternative. In addition, there would be impacts from providing the needed level of training. In general, the Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would require the most labor to implement (due to the combination of a long period of maintaining stored materials plus processing activity). The Continuing Storage and Processing to (plutonium) Metal Alternatives would involve activities similar to those performed in the past; as a result, facility personnel would have existing training and qualification programs to maintain core competency. The Processing to (plutonium) Oxide, Vitrification, and Improving Storage Alternatives would require additional levels of training; the only impact anticipated from such additional training would be the incremental funding and time required. TE

2.5.6 AGING FACILITIES

To some extent all the alternatives would involve the use of existing facilities, the ages of which vary widely. The canyons and reactor disassembly basins are more than 40 years old. Many of the other facilities in which DOE stores nuclear materials are between 30 and 40 years old. The newest facility, HB-Line, is less than 10 years old, but it is on top of the 40-year-old H-Canyon. Even though DOE has maintained these facilities since their construction, they contain equipment and systems that have become degraded because of their age and changes in mission. In some cases the degraded condition of equipment can pose operational limitations. For example, at one time the H-Canyon contained equipment that provided the capability to dissolve not only aluminum-clad reactor fuel but also fuel clad in stainless steel. The electrolytic dissolver used for this purpose is no longer functional and has been abandoned in place. As described in Section 2.5.1, several alternatives would involve the removal of obsolete or abandoned equipment so DOE could use the space for new equipment or facilities.

L11-17

Because of the ages of the facilities, they do not satisfy all current DOE requirements for the design and construction of nuclear facilities. For example, the canyons and associated B-Line facilities were built (during the Cold War when a primary concern was the potential for an attack) to resist a large external blast. The blast-resistant features of the canyons also make them resistant to such external natural phenomena as tornadoes and earthquakes. However, the canyons were not designed to withstand a severe earthquake (defined as producing a lateral ground acceleration that is 20 percent that of gravity or 0.2*g*), as they would be if DOE were to build them today. Another example concerns the reactor disassembly basins. These water-filled basins were built of concrete without a lining for the walls and floor. The bare concrete surfaces make it difficult to control the water chemistry in the basins. In addition, the basins do not have systems or equipment to detect leaks, which could contaminate the surrounding soil and groundwater. Rather, the primary means of detection is a surrounding network of sampling wells that monitor groundwater quality near the facility. Similar facilities constructed to today's standards would not have these vulnerabilities, which DOE has acknowledged in several recent studies (DOE 1994d,e).

The continued use of or reliance on these facilities to store or process nuclear materials is an important factor for DOE consideration. Because the facilities do not meet current design and construction requirements, a facility-related vulnerability could produce environmental impacts. As discussed above, the canyons would not maintain structural confinement of nuclear materials in a severe earthquake. The estimates of potential environmental consequences from accidents took this acknowledged vulnerability into consideration. For example, Tables E-4 through E-12 in Appendix E indicate that many existing facilities probably would not maintain confinement of nuclear materials during a severe earthquake. These tables list resulting releases of nuclear material (in curies) and potential consequences (dose) to nearby workers and the public. Continuing with this example, if DOE were to design and construct new facilities, there would be no environmental consequences from a severe earthquake because such a vulnerability would not exist.

L11-17 Similarly, DOE considered other types of facility vulnerabilities in estimating the potential consequences from the accidents discussed in Appendix E. Some examples are (1) a fire that could spread in a facility until it breached containers of nuclear material due to a lack of detection or extinguisher systems, (2) systems that cool nuclear materials stored in tanks that could leak and transfer such material outside the facility before detection, or (3) piping configurations in the canyons that personnel could use inadvertently to transfer solutions of nuclear material to an outside facility tank where they could overflow or spill.

DOE has conducted many reviews to evaluate facility vulnerabilities and has assessed its facilities for compliance with current requirements. DOE has also analyzed the effect on workers and the public from normal and potential accident conditions which could result from operation of facilities with these vulnerabilities. The analysis work was accomplished as a part of ongoing safety review programs and is separate from the NEPA process. Such impact information is represented in this EIS. The analysis of impacts has, in some cases, prompted DOE to take corrective action based on potential impact alone. For example, DOE has disconnected some tanks of radioactive solutions in the canyons from the canyon cooling system and has isolated canyon tanks by removing interconnected piping to preclude leaks or an inadvertent transfer which could result in a release of radioactive material outside the canyon. In other cases, the derived impact was determined to be small and not to warrant actions beyond those which could be taken using existing facilities, equipment, and personnel. For example, one vulnerability common to many facilities is that the facility could sustain structural damage in the event of a severe earthquake. This type of earthquake has been estimated to occur once every several thousand years. It would be expensive to modify facilities so no structural damage would occur from this accident. Rather, DOE mitigates the consequences of such accidents (to be no greater than the impacts represented in this EIS) using engineering safeguards, such as structurally reinforcing tanks, and administrative controls, such as limiting the amount of radioactive material that can be contained in a facility.

A DOE objective is the ultimate removal of nuclear materials from many of these facilities, such as the reactor disassembly basins and canyons, which are no longer required for national defense. The removal of nuclear materials would reduce the risks associated with facility vulnerabilities. For example, if the nuclear materials were removed and only residual levels of radioactive contamination remained, the consequences from a severe earthquake would be greatly reduced, minimizing public and worker exposures. After removal of the nuclear materials, the facilities would be available for decontamination and decommissioning. All the alternatives discussed in this EIS except No Action would support DOE's ultimate objective of removing nuclear materials from these facilities and proceeding with decontamination and decommissioning. L11-17

2.5.7 MINIMUM CUSTODIAL CARE

The vitrification alternatives would eventually result in a stabilized form of material that would require a minimum of custodial care. However, continued custodial care of the materials would be required in canyons, vaults, or high-level waste tanks until vitrification had been accomplished. Continuing Storage would involve maintaining candidate materials for stabilization (necessitating increasing surveillance, maintenance, and corrective actions) for the longest time and, therefore, can be considered the least advantageous alternative in this regard.

Other processing and improving storage alternatives would have varying levels of custodial care requirements. Stable materials would need less care than candidate materials for stabilization, so the preferred alternatives would involve less custodial care than other alternatives because they would stabilize the materials the earliest.

2.6 Cost

To determine the potential effect of integrating various combinations of alternatives into an overall stabilization and management program at the SRS, DOE estimated the costs over the next 10 years to implement each of the four management scenarios described in Section 2.5.3. The estimates include both operating and capital (i.e., construction) costs (WSRC 1995b). Figures 2-7 through 2-10 show L11-11 L11-12

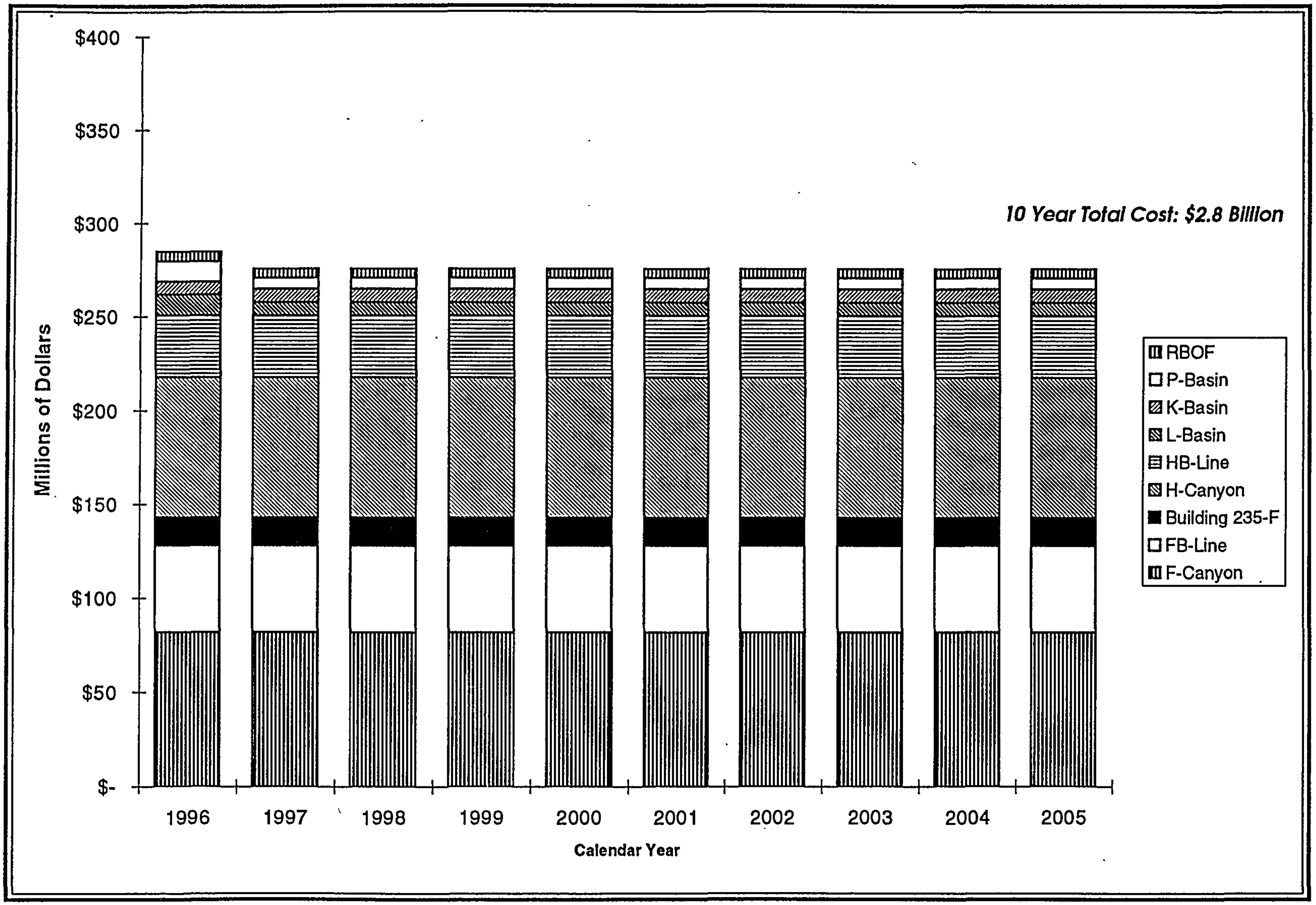


Figure. 2-7. 10-Year Cost Profile - No-Action Scenario.

TC

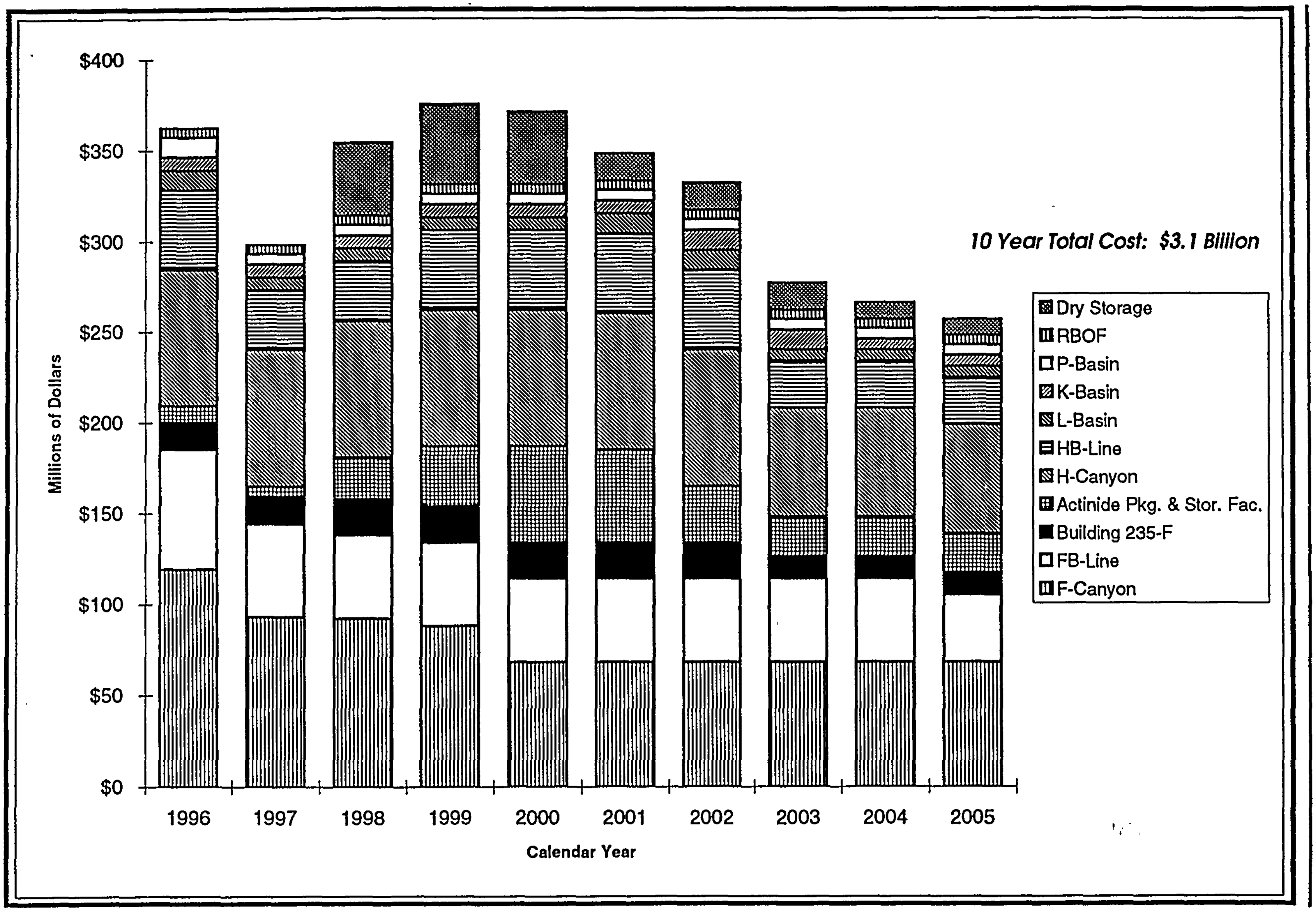


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Figure 2-8. 10-Year Cost Profile - Minimum Processing Scenario.

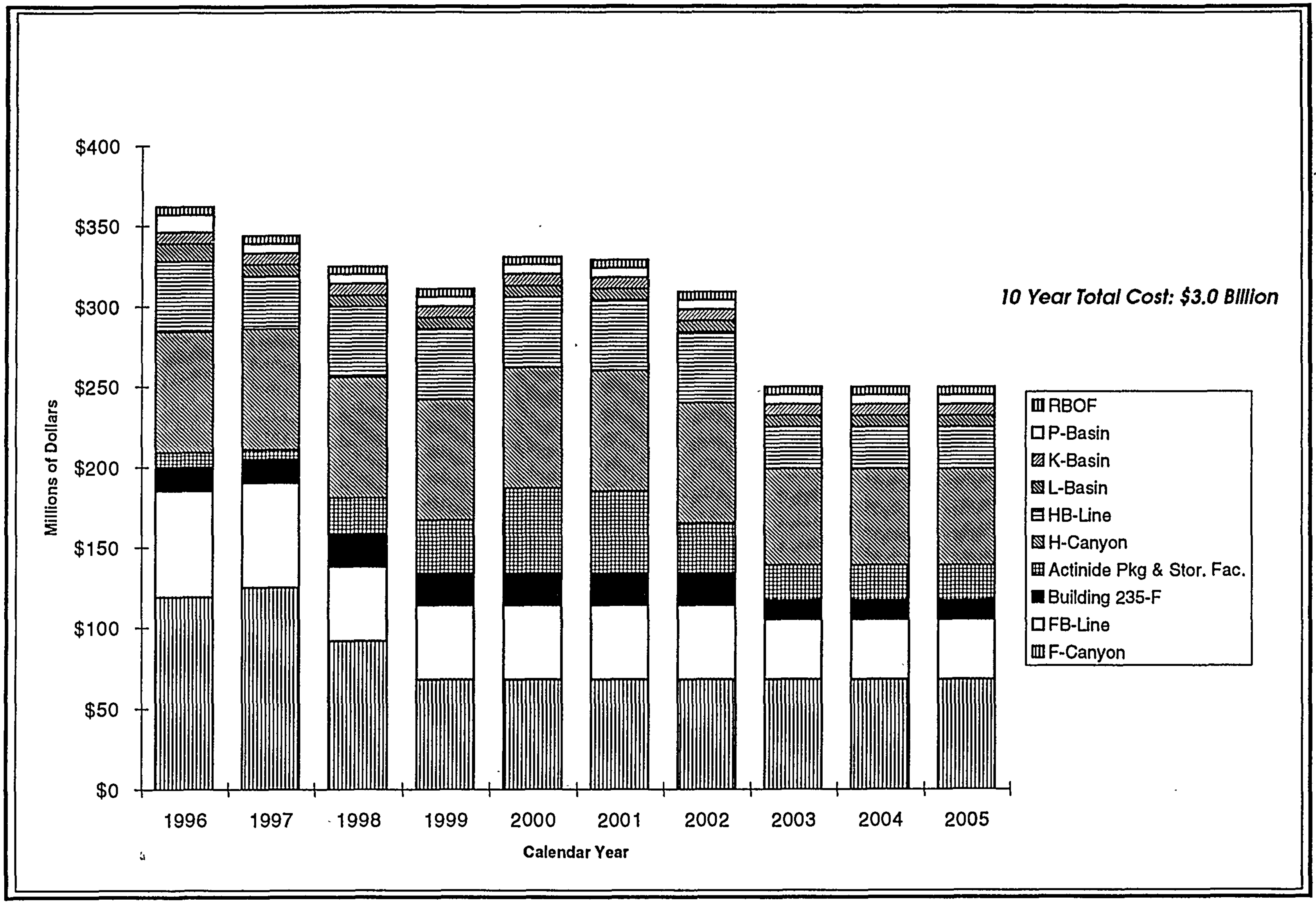

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Figure 2-9. 10-Year Cost Profile - Preferred Scenario.

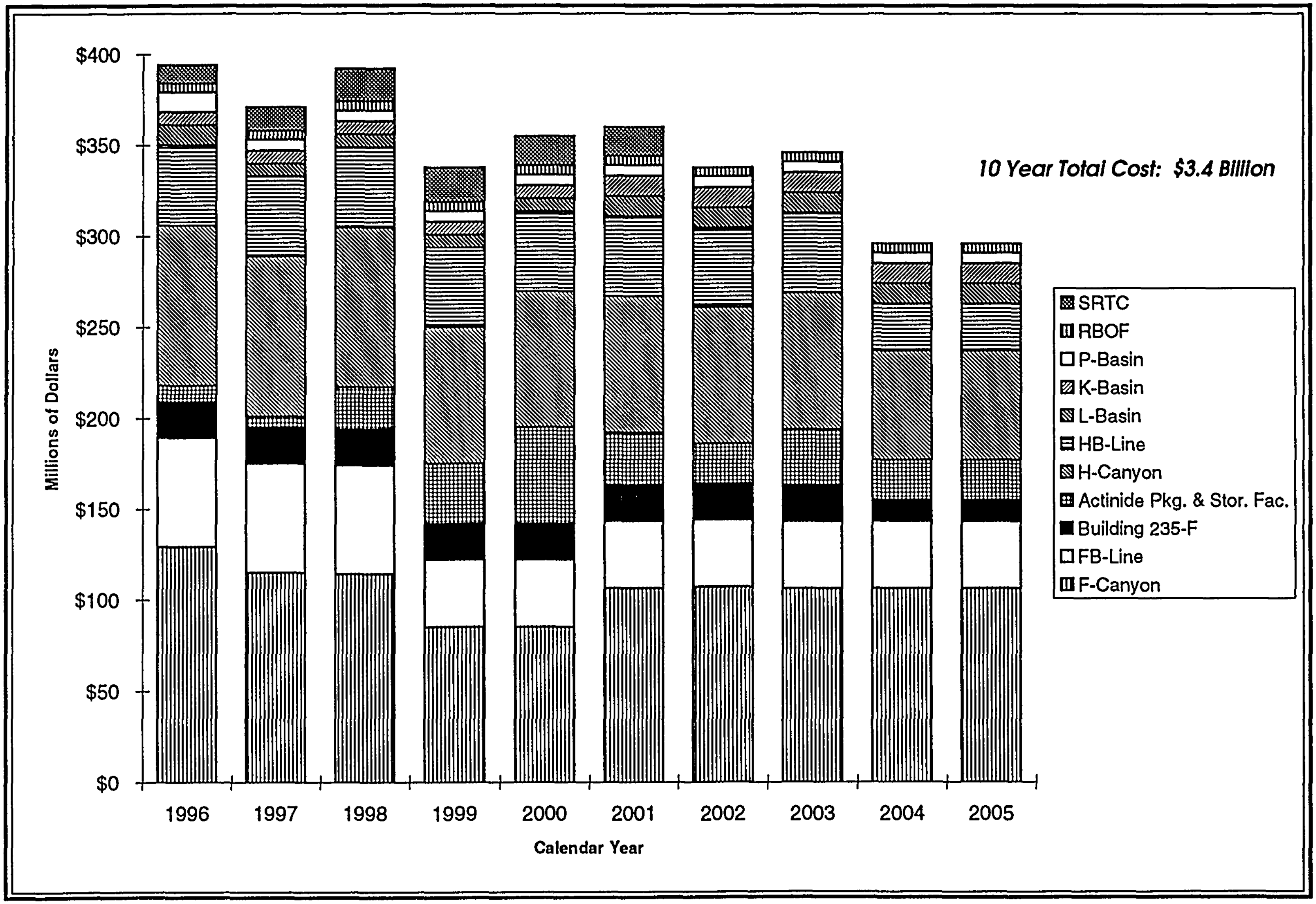

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Figure 2-10. 10-Year Cost Profile - Comparative Scenario.

the 10-year cost profiles for each scenario. These profiles include the total and individual costs required each year to manage and stabilize the nuclear materials.

The cost profiles illustrate several important points. First, as shown by Figure 2-7, DOE estimates it would require $2.8 billion dollars to continue management of the nuclear materials in their present locations and storage conditions. As explained in Section 2.5.3, few SRS facilities have enough storage capacity to support significant consolidation of the nuclear materials. The variety of physical and chemical forms of the nuclear materials further limits DOE's ability to consolidate storage and thereby reduce future operating costs. In other words, without taking action to alter the current physical and chemical form of many nuclear materials (Programmatic and Candidates for Stabilization) or to build new facilities (e.g., Actinide Packaging and Storage Facility, Dry Storage Facility), DOE would have to maintain the materials indefinitely in a wide variety of physical and chemical forms and in facilities that are generally over 40 years old.

The Preferred Scenario and Minimum Processing Scenario are representative of alternative combinations that DOE is most likely to choose. As Figures 2-8 and 2-9 show, the total cost of the 10-year management program under these two scenarios would be approximately $3.0 billion. Under both scenarios, DOE would chemically stabilize existing solutions containing nuclear material.
L11-11
L11-12 The resulting metal, oxides, and glass forms of nuclear material would be consolidated in a new Actinide Packaging and Storage Facility (see Figures 2-4 and 2-5). Under the Preferred Scenario, DOE would use a variety of alternatives to stabilize plutonium vault materials, whereas under the Minimum Processing Scenario, DOE would use only thermal stabilization and repackaging. However, both scenarios would involve the construction and operation of a new Actinide Packaging and Storage Facility, and Figures 2-8 and 2-9 reflect the associated costs.

Although the differences in estimated total cost over the 10-year period would be very small between the Preferred and Minimum Processing Scenarios, the management approach used for aluminum-clad materials stored in basins would be markedly different. Under the Preferred Scenario, DOE would chemically stabilize the Mark-31 targets, failed Taiwan Research Reactor fuel, and declad Experimental Basin Reactor-II slugs stored in failed canisters. All other basin materials that are Candidates for Stabilization would remain in wet storage until DOE selected a preferred method for stabilization. Under the Minimum Processing Scenario, all materials that are Candidates for Stabilization would be heat-treated, packaged, and placed in dry storage as soon as a Dry Storage Facility could be designed and constructed. Figure 2-9 shows that the major costs for constructing a Dry Storage Facility (on an accelerated schedule) would occur in the 1998-to-2000 period. The costs of a new Dry Storage Facility would result in slightly higher total costs to DOE during the years of construction and initial operation (i.e., until the materials could be transferred from the basins and placed in dry storage). However, after the materials had been placed in dry storage, the annual costs to DOE would be reduced by 2003 to levels similar to those for the Preferred Scenario.

Under either of the two scenarios, reductions in annual costs would begin to occur as radioactive solutions, vault materials, and basin materials were stabilized or converted to potentially usable forms. After the stabilization of radioactive solutions and materials in canyon basins, DOE could begin deactivation of the F- and H-Canyons. After the construction of the proposed Actinide Packaging and Storage Facility, DOE could consolidate the storage of all vault materials at the SRS. DOE could then begin deactivation of the existing vaults and B-Lines. Under the Minimum Processing Scenario, DOE could begin deactivation of the reactor basins as soon as the materials were transferred to dry storage. Under any scenario, DOE would transfer material from the P-Reactor basin to another reactor basin. DOE could begin deactivation of the P-Reactor on removal of the basin material. Any decisions by DOE to deactivate an SRS facility would depend on the outcome of ongoing programmatic studies (see Sections 1.6 and 2.9).

DOE expects costs associated with operating and maintaining the facilities to decrease as nuclear materials are stabilized and consolidated for interim management. In developing the cost profiles, DOE included estimates of the annual costs to maintain the nuclear facilities in an operable state after the removal of all but residual levels of radioactive material (i.e., only contaminated equipment and structures remain). The profiles do not reflect further reductions in annual costs that DOE expects after decisions to deactivate the facilities. DOE expects significant reduction in annual costs as and if facilities are deactivated and is continuing its investigation of these potential savings to assist in the decision process (see Section 2.9). However, substantial costs could be required to decontaminate and decommission such major nuclear facilities. DOE believes the associated deactivation, decontamination, and decommissioning costs are beyond the scope of this EIS and that such a discussion would be premature in light of ongoing programmatic studies that could involve the future use of SRS facilities. L11-11 L11-12

As discussed above, although the Comparative Scenario might be representative of thousands of other possible combinations, DOE probably would not choose it for management of the materials. However, because DOE developed the Comparative Scenario to illustrate the effects of the combination of alternatives with the greatest impacts, it involves alternatives requiring extended periods to perform technical studies and chemically stabilize materials (see Figure 2.6). The extended periods also indicate increased costs. For example, Figure 2-10 shows incremental Savannah River Technology Center costs (1996 to 2001) to perform technical studies in support of dissolving and transferring reactor basin materials directly to the SRS high-level waste system for eventual vitrification in the Defense Waste Processing Facility. As Figure 2-6 shows, the time required to perform the technical studies and demonstration work would extend the time nuclear materials would have to be stored and processed in the reactor basins and canyons, respectively. Therefore, in many ways the Comparative Scenario illustrates not only the combination of alternatives with the greatest impacts, but also one of the most costly combinations.

As indicated in Chapter 4, each alternative would produce radioactive waste in varying amounts. Table 2-15 lists estimated costs to manage generated wastes (by type) for each scenario. For each scenario, the costs include vitrification of high-level liquid waste in the Defense Waste Processing Facility or solidification in the Saltstone facility (Appendix C describes these facilities), even though this action would not occur within the 10-year period. DOE included vitrification and solidification costs for the high-level waste because they represent a potentially significant fraction of the long-term cost for each scenario.

Table 2-15. Estimated costs to manage generated wastes (millions of dollars).[a]

L11-11
L11-12

	Scenario			
Waste type	No Action	Minimum Processing	Preferred	Comparative
High-level liquid	$180	$280	$330	$2,300
Solid waste[b]	$190	$250	$220	$240
Total	$370	$530	$550	$2,500

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a. Adapted from WSRC (1995b).
b. Solid waste includes transuranic, mixed, and low-level waste.

2.7 Preferred Alternatives

In selecting a preferred alternative for each material category, DOE considered environmental impacts, duration of stabilization activities, cost, and other factors, as discussed in Sections 2.3, 2.4, 2.5, and 2.6. Chapter 4 contains detailed information on environmental impacts.

A key factor that DOE used in selecting the preferred alternatives was Defense Nuclear Facilities Safety Board Recommendation 94-1 to the Secretary of Energy (DNFSB 1994). As described in Chapter 1, the Board recommended that DOE accelerate actions to convert certain nuclear materials to forms or conditions suitable for safe interim storage. The Secretary of Energy accepted the DNFSB recommendation and in response submitted an Implementation Plan (DOE 1995a) that outlines proposed corrective actions. The DNFSB recommendation contained specific objectives related to the completion of actions. The following objectives were related to the SRS materials:

- Convert the dissolved plutonium and transplutonium isotopes (i.e., americium and curium solutions) in tanks in the F-Canyon to safer forms for interim storage within 2 to 3 years.

- Expedite preparations to process the deteriorating irradiated reactor fuel stored in the reactor basins to a form suitable for safe interim storage (within 2 to 3 years) until DOE selects an option for ultimate disposition.

- Expedite preparations to repackage the plutonium metal that is in contact with, or in proximity to, plastic or to eliminate the associated existing hazard in any other way that is feasible and reliable. Storage of plutonium materials generated through this remediation process should be such that there would be no need to open the containers for additional treatment for a reasonably long time. The project plan should provide that, within a reasonable period of time (such as 8 years), the storage of plutonium metal and oxide should conform to the DOE Standard on storage of plutonium.

"The Department has broadened the scope of the response to Recommendation 94-1 to include additional bulk liquids and solids containing fissile materials and other radioactive substances in spent fuel storage pools, reactor basins, reprocessing canyons, processing lines and various facilities which require conversion to forms, or establishing conditions, suitable for safe interim storage. The scope was broadened to ensure that similar materials under similar conditions receive the same degree of management attention as those noted by the Board in its recommendation" (DOE 1995a). Consistent with the above, the nuclear materials identified in this EIS categorized as either "Programmatic" or "Candidates for Stabilization" were included in the Department's commitment to achieve the objectives identified in the Board recommendation, including the Board's objectives for timeframes for converting or repackaging the materials into forms safe for interim storage. L11-1 L11-2

The following sections identify the DOE preferred alternative and summarize the performance of the preferred alternative for each material category. In this Final EIS DOE has added some of the Taiwan Research Reactor fuel and Experimental Breeder Reactor-II slugs currently stored in the Receiving Basin for Offsite Fuel as material in the "Candidate for Stabilization" category. As a result, DOE has included Section 2.7.11 to discuss the preferred alternative for stabilizing this material. DOE had classified this material as Stable in the Draft EIS. However, questions raised by the Defense Nuclear Facilities Safety Board (DNFSB 1995) and comments received on the Draft EIS resulted in a

L11-1 L11-2 reevaluation by the Department. DOE concluded the same problems exist for some of the TRR fuel and EBR-II slugs as for the Mark-31 targets (i.e., the primary containment no longer provides a barrier to prevent the release of fission products from the material and does not isolate the material from storage basin water).

2.7.1 STABLE MATERIAL

DOE believes that material in this category is suitable for continued storage and that no actions are necessary to meet the purpose or need for this EIS. Therefore, the preferred alternative for the material in this category is Continuing Storage (i.e., No Action). Under this alternative, DOE would manage the material in its existing form to maintain the health and safety of workers and the public.

2.7.2 PLUTONIUM-242

L2-1 L2-4 L11-1 DOE has selected Processing to Oxide as the preferred alternative for plutonium-242. This alternative would have health impacts comparable to those of the other alternatives analyzed for both normal operations and potential accidents associated with facility operations. The health impacts for transportation accidents would be comparable to those for other processing alternatives but slightly higher than those for the No-Action and Improving Storage Alternatives. Impacts to air and water resources would be similar to those from the other alternatives. Utilities consumption for the Processing to Oxide Alternative would be comparable with that for the other alternatives and the lowest for electricity consumption. The preferred alternative would have the smallest estimated impact in terms of waste generation.

DOE could accomplish the Processing to Oxide Alternative in about 6 months. DOE has the facilities, equipment and personnel needed to convert the material to an oxide, and the oxide form would meet the potential programmatic need. The time to accomplish the other stabilization alternatives would range from about 3-1/2 years to more than 20 years. The cost to accomplish the Processing to Oxide Alternative would be comparable to the other stabilization alternatives and would be lower than all but Processing and Storage for Vitrification in the Defense Waste Processing Facility.

The Processing to Oxide Alternative would produce a material that is less proliferation resistant than the products of the other alternatives except Processing to Metal. The oxide form would require less custodial care than leaving the material in solution form but more care than alternatives that would result in a vitrified product; this is because of the range of surveillance and periodic inspection requirements associated with the final material forms from the various alternatives. The Processing to Oxide Alternative would be comparable to the other alternatives in terms of the age of the facilities used over 10 years.

2.7.3 AMERICIUM AND CURIUM

DOE has selected Vitrification (F-Canyon) as the preferred alternative for americium and curium. This alternative would have health impacts comparable to those from the other alternatives for both normal operations and potential accidents associated with facility operations. Health impacts from potential transportation accidents would be comparable to those from other processing alternatives. In addition, impacts to air and water resources would be similar to those from the other alternatives. Utilities consumption for the Vitrification (F-Canyon) Alternative would be comparable to that for the other alternatives; however, this alternative would have the lowest electricity consumption. This alternative would be similar to the other alternatives in terms of waste generation (e.g., somewhat higher than Continuing Storage but significantly lower than Processing to Oxide).

DOE could accomplish the Vitrification (F-Canyon) Alternative in about 4 years by modifying F-Canyon to install vitrification equipment. The time to accomplish the other stabilization alternatives would range from about 6 years to more than 20 years. The cost to accomplish the Vitrification (F-Canyon) Alternative would be comparable to that for the other stabilization alternatives (e.g., the modification of F-Canyon for either the vitrification or oxide alternative would have roughly the same cost and would require the same time for completion). L2-1 L2-4 L11-1

The glass forms produced by the Vitrification (F-Canyon) Alternative would require the least amount of custodial care because such forms would be essentially inert. The Vitrification (F-Canyon) Alternative would be comparable to the other alternatives in terms of the age of the facilities used over 10 years.

2.7.4 NEPTUNIUM-237

DOE has selected Processing to Oxide as the preferred alternative for neptunium-237. The health impacts produced by this alternative would be among the lowest of the alternatives analyzed for accidents associated with facility operations. Health impacts for normal operations and transportation accidents would be the highest in comparison to the other alternatives. Impacts to air and water resources would be similar to those from the other alternatives. Utilities consumption for the

Processing to Oxide Alternative would be comparable to that for the other alternatives. This alternative would have the highest waste generation of all the alternatives.

DOE could accomplish the Processing to Oxide Alternative in about 3 years. DOE has the required facilities, equipment, and personnel to convert the material to an oxide, which would meet the potential programmatic need. The time to accomplish the other stabilization alternatives would range from about 4 years to more than 20 years. The cost to accomplish the Processing to Oxide Alternative would be comparable to that for the other stabilization alternatives; No Action would be the costliest overall.

The oxide form would require less custodial care than leaving the material in solution form but would require more care than alternatives that would result in a vitrified product. The custodial care would be principally the surveillance and periodic inspection requirements associated with the final material forms from the various alternatives. The Processing to Oxide Alternative would be comparable to the other alternatives in terms of the age of the facilities used over 10 years. DOE has selected Continued Storage (No Action) as the preferred alternative for targets that contain americium and curium.

2.7.5 H-CANYON PLUTONIUM-239 SOLUTIONS

L11-1

DOE has selected Processing to Oxide as the preferred alternative for the H-Canyon plutonium-239 solutions. This alternative would have health impacts comparable to those from the other alternatives for both normal operations and potential accidents associated with facility operations. Health impacts for potential transportation accidents would be among the highest. Impacts to air and water resources would be similar to those from the other alternatives. Utilities consumption for the Processing to Oxide Alternative would be comparable to that for the other alternatives; however, this alternative would have the lowest electricity use. This alternative would have the smallest impact in terms of waste generation.

DOE could accomplish the initial oxide conversion for the Processing to Oxide Alternative in 1 year. When the heating and packaging facility became available, the final packaging operation would take about 2 months. DOE has the facilities, equipment and personnel needed to convert the material to an oxide. The time to accomplish the other stabilization alternatives would range from about 4 years to more than 20 years. The cost to accomplish the Processing to Oxide Alternative would be considerably lower than that for the other alternatives.

The Processing to Oxide Alternative would produce a material that is less proliferation resistant than the products of the other alternatives except Processing to Metal. The oxide form would require less custodial care than leaving the material in solution form but more care than alternatives that would result in a vitrified product; this is due to the range of surveillance and periodic inspection requirements associated with the final material forms from the various alternatives. The Processing to Oxide Alternative would be comparable to the other alternatives in terms of the age of the facilities used over 10 years.

2.7.6 H-CANYON URANIUM SOLUTIONS

DOE has selected Blending Down to Low Enriched Uranium as the preferred alternative for the H-Canyon uranium solutions. This alternative would have health impacts comparable to those from the other alternatives for both normal operations and potential accidents associated with facility operations. The health impacts for potential transportation accidents would be comparable to those from other processing alternatives. Impacts to air and water resources would also be similar to those from the other alternatives. Utilities consumption for Blending Down to Low Enriched Uranium would be comparable to that from the other alternatives; however, this alternative would be among the lowest for electricity consumption. This alternative would have waste generation impacts that would be similar to those from the other alternatives (e.g., higher than Processing to Oxide and lower than No Action). L11-1

DOE could accomplish the Blending Down to Low Enriched Uranium Alternative in about 2 years. DOE has the facilities, equipment, and personnel needed to convert the material to low-enriched oxide. Some facility modifications would be required to support the transfer of the material from H-Area to FA-Line. The time to accomplish the other stabilization alternatives would range from about 4 years to more than 20 years. Blending Down to Low Enriched Uranium would have a significantly lower cost than the other alternatives.

The Blending Down to Low Enriched Uranium Alternative would eliminate the proliferation concern for the existing material. The oxide form would require a low level of custodial care (e.g., monitoring a warehouse). This alternative would be comparable to the other alternatives in terms of the age of the facilities used over 10 years; however, it would use equipment about 20 years older than equipment that would be used for Processing to Oxide in the Uranium Solidification Facility.

2.7.7 PLUTONIUM AND URANIUM STORED IN VAULTS

DOE has selected Improving Storage, Processing to Metal, Processing to Oxide, and Vitrification (F-Canyon) as the preferred alternatives for the plutonium and uranium stored in vaults. DOE believes that about half the containers currently used to store material hold plutonium and uranium for which Improving Storage would be applicable. The material in the remaining containers would be stabilized by the Processing to Metal, Processing to Oxide, or Vitrification (F-Canyon) Alternative. In general, DOE would use the Processing to Metal Alternative for material which did not contain significant quantities of uranium-235 until the other stabilization alternatives could be implemented.

The Processing to Oxide Alternative would be used for stabilizing plutonium-238, for materials with plutonium-239 containing uranium-235, and for parallel processing efficiency. The Vitrification (F-Canyon) Alternative would be used if the technology was demonstrated successfully and providing the vitrification facility was constructed through the selection of the Vitrification (F-Canyon) Alternative for americium and curium solutions. These alternatives would have health impacts comparable to those for the other alternatives for both normal operations and potential accidents associated with facility operations. Health impacts for potential transportation accidents and impacts to air and water resources would be similar for all alternatives. Utilities consumption for the L11-1 Vitrification (F-Canyon), Processing to Metal, and Processing to Oxide Alternatives would be comparable to that for the other alternatives. Improving Storage would have the lowest electricity consumption. The Improving Storage Alternative would generate no high-level waste, and the waste generation rates for Processing to Oxide, Processing to Metal, and Vitrification (F-Canyon) would be similar to those for the other stabilization alternatives except Processing and Storage for Vitrification in the Defense Waste Processing Facility, which would have a high volume of high-level waste. The volume of transuranic, hazardous and mixed, and low-level waste would be comparable for all the alternatives.

The Improving Storage, Processing to Metal, Processing to Oxide, and Vitrification (F-Canyon) Alternatives would take about 7-1/2, 4, 4, 5-1/2, and 5-1/2 years, respectively, to complete. During the first 2 years of the Improving Storage Alternative, DOE would complete the characterization of all material; material that DOE could not stabilize by repackaging would have been identified and stabilized or scheduled for stabilization as a part of the Processing to Metal, Processing to Oxide, or Vitrification (F-Canyon) Alternative. DOE would have the facilities and personnel needed to perform inspection activities and to convert the material to a metal or oxide. However, DOE would have to install additional equipment to support packaging material to meet the plutonium storage standard criteria (DOE 1994h) and to vitrify the material in F-Canyon. The time to accomplish the other stabilization alternatives would range from about 5-1/2 years to more than 20 years. The Improving Storage Alternative would have the lowest implementation cost of all the alternatives and Processing and Storage for Vitrification in the Defense Waste Processing Facility would have the highest.

The Improving Storage, Processing to Metal, and Processing to Oxide Alternatives would not significantly alter the proliferation resistance status of this material, which is in metal and oxide forms. Vitrification (F-Canyon) and Processing and Storage for Vitrification in the Defense Waste Processing Facility would produce the most proliferation-resistant form of the material. The preferred alternatives would require the Site to maintain current levels of custodial care, which would be the highest levels among the alternatives because of the range of surveillance and periodic inspection requirements associated with the final material forms produced by the alternatives. The preferred alternatives would be comparable to the other alternatives in terms of the age of the facilities used over 10 years. However, the preferred alternatives would require new processing and packaging equipment.

2.7.8 MARK-31 TARGETS

DOE has selected Processing to Metal as the preferred alternative for the Mark-31 targets. This alternative would have health impacts comparable to those from the other alternatives for both normal operations and potential accidents associated with facility operations. The health impacts for transportation accidents would be comparable to those from other alternatives that would accomplish stabilization, but would be higher than the No-Action, Processing and Storage for Vitrification in the Defense Waste Processing Facility, and Improving Storage Alternatives. Impacts to air and water resources and waste generation volumes would be similar to those from the other processing alternatives. Utilities consumption for the Processing to Metal Alternative would be comparable to that from the other alternatives; however, this alternative would be among the highest for electricity consumption. L11-1

DOE could accomplish the Processing to Metal Alternative in about 3 years. Initial stabilization would occur within the first year by dissolving the materials and converting the plutonium to metal. The metal would be packaged to satisfy DOE storage requirements (DOE 1994h) within 3 years. DOE has the facilities, equipment, and personnel needed to convert the material to a metal. The time to accomplish the other alternatives would range from about 3-1/2 years to more than 20 years. The accelerated schedule for the alternative that involves dry storage would require more than 5 years to complete all operations, which would be at least 2 years longer than the schedule for the Processing to Metal Alternative. The cost to accomplish the preferred alternative would be similar to, and the lowest cost of, the stabilization alternatives. The cost for construction and operation of a dry storage facility could be as low as $78 million but would still be more than twice that of the preferred alternative.

The Processing to Metal Alternative would produce a material that would be the least proliferation resistant of the alternatives. However, the amount of plutonium-239 contained in the targets is a very small fraction of the existing DOE inventory of weapons-usable plutonium metal. In addition, DOE does not believe the Mark-31 targets contain enough plutonium to increase proliferation concerns, regardless of the stabilization alternative. DOE has committed to not use plutonium-239 separated as a result of the stabilization, phaseout, shutdown, or cleanout of weapons complex facilities for nuclear explosives purposes.

DOE proposes to stabilize the Mark-31 targets at this time for several reasons. This alternative fully alleviates the problems created by the current material storage configuration and would accomplish stabilization in the period recommended by the DNFSB. The material has failed cladding, exposing uranium metal to the storage basin water. The uranium metal is oxidizing on contact with water and this reaction releases entrained fission products. The reaction occurs independently of the basin water purity (i.e., conductivity) and will continue as long as the uranium metal is in contact with water because the uranium metal oxidation rate is not sensitive to water purity. Adjusting the pH of the basin water to be more basic could slow the reaction, but basic pH levels would increase the rate of cladding corrosion on other material in the basin. Stabilization of the Mark-31 targets at this time could occur in conjunction with stabilization of other nuclear material stored in F-Canyon and FB-Line. This integrated approach would help minimize stabilization program costs at the SRS without extending the operating life of either F-Canyon or FB-Line.

L11-1

By processing the Mark-31 targets to metal, DOE would use existing processes and technologies. The technical uncertainty would be low and the associated costs would be well established.

DOE also proposes to use the Processing to Metal Alternative to manage the 34 intact Mark-31 targets stored in the Receiving Basin for Offsite Fuel. DOE believes that the targets are intact and that they do not pose a risk. They represent an extremely small amount of material in addition to the approximately 16,000 Mark-31 targets that would require stabilization by Processing to Metal. If DOE were to maintain these 34 targets in continued storage (i.e., No Action), they would ultimately require long-term management and disposition.

2.7.9 MARK-16 AND -22 FUEL

DOE's preferred alternative for the Mark-16 and -22 spent fuel is Continuing Storage (No Action). No Action is recommended at this time because DOE has elected to perform a further review of costs, schedules, and the technical uncertainty of dry storage techniques for failed fuel before making a decision on management options. The additional review was prompted by public comments that DOE received on potential alternatives to technologies involving chemical dissolution. As a consequence, DOE's preferred alternative for these fuels is Continuing Storage until the additional review is completed. DOE expects to complete the review within the next 2 to 4 months. DOE does not expect the additional comparison to cause substantial alterations of the estimates of environmental impacts for the alternatives being considered. DOE believes that this additional review for the Mark-16 and -22 spent fuel furthers the intent and purposes of the National Environmental Policy Act by ensuring that the decisionmaker receives sufficient information on which to base a decision.

Stabilization of these fuels by chemical dissolution could not begin for at least 1 year due to the current availability of resources and the operating status of candidate facilities for the materials at the SRS. Similarly, stabilization actions for alternatives involving dry storage of these materials could not begin for at least 5 years until the appropriate facilities have been designed and constructed. Therefore, the fuels would have to be stored for at least 1 year in their current location (i.e., No Action), regardless of the ongoing additional review. L11-1

However, DOE does not consider it to be in the best interest of the public or DOE workers to defer the issuance of this EIS for another 2 to 4 months simply to identify and select a preferred stabilization alternative for the Mark-16 and -22 fuels. The benefits derived from proceeding with stabilization decisions on all the other materials outweigh any benefits that might be derived from further delaying the NEPA process to select a preferred stabilization alternative for the Mark-16 and -22 fuels. Therefore, DOE has chosen Continuing Storage (No Action) as the preferred alternative for the immediate future so it could make stabilization decisions for the other materials on the basis of this Final EIS.

For the Mark-16 and -22 fuels, estimated waste generation volumes for the Blending Down to Low Enriched Uranium Alternative would be similar to those from the other alternatives. Processing and Storage for Vitrification in the Defense Waste Processing Facility would have the greatest overall estimated waste volume impact. With the exception of the Improving Storage Alternative (i.e., dry storage), the number of equivalent DWPF canisters (see Section 2.4) represents the volume of material intended for placement in a geologic repository after the completion of stabilization activities. Under the Improving Storage Alternative, DOE would add the number of dry storage canisters required to contain all the spent fuel to the number of equivalent DWPF canisters. If the spent fuel were counted as Material Test Reactor Elements, the estimated number of canisters intended for the proposed geologic repository could increase by a range from about 55 to 6,600. The large range is due to the uncertainty associated with the method DOE would use for criticality control. The lower value reflects heavy reliance on neutron poisons; the higher value reflects the use of restricted fissile material concentrations [i.e., 700 grams (25 ounces) equivalent of uranium-235 per canister; WSRC 1995a]. The range of canisters is most significant in terms of cost. The repository fee for 55 canisters would be about $50 million, while the fee for 6,600 canisters would be more than $2 billion. Before the spent nuclear fuel could be shipped to a geologic repository, some conditioning activity would be required. In addition, a new facility would be required because the conditioning activity would not occur for 35 years or more. Conditioning could involve operations ranging from opening the fuel canisters and adding a neutron poison to processing the fuel to blend it down or poison it before loading in a repository canister.

DOE could accomplish the Blending Down to Low Enriched Uranium Alternative in about 4-1/2 years. DOE has the facilities, equipment, and personnel needed to convert the material to an oxide. The time to accomplish the other stabilization alternatives would range from about 6 years to more than 20 years. The accelerated schedule for the dry storage alternative would require more than 7 years to complete all operations, which would be at least 2 years longer than the schedule for the Blending Down to Low Enriched Uranium Alternative. The cost for construction and operation of a dry storage facility could be as low as $131 million, but would still be more than the cost for any other alternative. The Blending Down to Low Enriched Uranium and Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternatives would have the lowest costs. L11-1

The Blending Down to Low Enriched Uranium Alternative would produce a material for which there would be no proliferation concerns. The low-enriched oxide form would require less custodial care than leaving the material in a reactor basin over the next 10 years and would require a similar level of care as alternatives that would result in a vitrified product; this would be due to the range of surveillance and periodic inspection requirements associated with the final material forms from the various alternatives. This alternative would be comparable to the other alternatives in terms of the age of the facilities used over 10 years; however, it would use equipment that is about 20 years older than equipment that would be used for Processing to Oxide in the Uranium Solidification Facility.

2.7.10 OTHER ALUMINUM-CLAD TARGETS

DOE's preferred alternative for the Other Aluminum-clad Targets is Continuing Storage (No Action). No Action is recommended at this time because DOE has elected to further review costs, schedules, and the technical uncertainty of dry storage techniques for this material. The additional review was prompted by public comments that DOE received on potential alternatives to technologies involving chemical dissolution. Consequently, DOE's preferred alternative for these targets is Continuing Storage (No action) until the additional review is completed. The review is expected to be completed within the next 4 to 6 months. DOE does not expect the additional comparison to substantially alter the estimates of environmental impacts for the alternatives being considered. DOE believes that additional review concerning alternatives for the targets furthers the intent and purposes of NEPA by ensuring that during the NEPA process federal agencies be as responsive as practicable to public comments and the decisionmaker is provided with sufficient information upon which to base a decision.

Stabilization of these fuels by chemical dissolution could not begin for at least 1 year due to current availability of resources and the operating status of candidate facilities for the materials at the SRS. Similarly, stabilization actions for alternatives involving dry storage of these materials could not begin for at least 5 years until the appropriate facilities have been designed and constructed. Therefore, the targets would be managed for at least 1 year by continuing to store them in their current location (i.e., No Action), irrespective of any stabilization alternative. L11-1

However, DOE does not consider it would be in the interest of the public or its workers to defer issuance of this EIS for another 4 to 6 months simply to identify and select stabilization alternatives for the Other Aluminum-clad Targets. The benefits derived from proceeding with stabilization decisions on all the other materials outweigh any benefits which might be derived from further delaying the NEPA process in order to select stabilization alternatives for the Other Aluminum-clad Targets. In light of this, DOE has chosen Continuing Storage as the preferred alternative so that stabilization actions for other materials can proceed.

The estimated impacts from potential facility accidents for the Continuing Storage Alternative and the Improving Storage Alternative (both schedules) would be essentially the same and would be the lowest of all the alternatives. The impacts would be essentially the same because the alternatives involve storage activities over the next 10 years. Impacts to air and water resources and impacts from utilities consumption would be similar among the alternatives which would accomplish stabilization of the material in the next 10 years.

In general, waste generation volumes for the Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would be higher than those for the other alternatives. With the exception of the Improving Storage Alternative (i.e., dry storage) the number of equivalent DWPF canisters (see Section 2.4) would represent the volume of material intended for placement in a geologic repository after the completion of stabilization activities. Under the Improving Storage Alternative, the number of dry storage canisters that would be required to contain all the material would be added to the number of equivalent DWPF canisters. If the targets were counted as Material Test Reactor Elements, the estimated number of canisters going to the proposed geologic repository would increase only slightly (above the number of waste canisters) because the amount of material would be small and the fissile material content of the material would be low. DOE could determine that the material would not be suitable for placement in a geologic repository after loading the targets for dry storage. In this case, additional processing work of some type would be required and additional high-level waste in the form of equivalent DWPF canisters would be generated.

The Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would transfer the material to the high-level waste tanks in about 6 years; however, stabilization would not occur for more than 20 years. DOE has the facilities, equipment, and personnel needed to perform this alternative; the cost to accomplish this alternative would be the least of all the alternatives. L11-1

The material would require essentially the same level of custodial care for all the alternatives because it would remain in a reactor basin for most or all of the next 10 years. All the alternatives would rely on the use of facilities of similar age.

2.7.11 TAIWAN RESEARCH REACTOR FUEL AND EXPERIMENTAL BREEDER REACTOR-II SLUGS

DOE has selected Processing to Metal as the preferred alternative for the TRR and EBR-II material. This alternative would have health impacts comparable to those from the other alternatives for both normal operations and potential accidents associated with facility operations. The health impacts for transportation accidents would be comparable to those from other alternatives that would accomplish stabilization but would be higher than the No-Action, Processing and Storage for Vitrification in the Defense Waste Processing Facility, and Improving Storage Alternatives. Impacts to air and water resources and waste generation volumes would be comparable to those from the other processing alternatives but higher than those for the alternatives involving storage. Utilities consumption for the

Processing to Metal Alternative would also be comparable to that from the other alternatives; however, this alternative would be among the highest for electricity consumption.

DOE has the facilities, equipment, and personnel needed to convert the material to a metal and could accomplish the Processing to Metal Alternative in no more than 3 years. Initial stabilization of the failed TRR fuel and EBR-II canister would occur within the first year by dissolving the materials and converting the plutonium to metal. The metal would be packaged to satisfy DOE storage requirements (DOE 1994h) within 3 years. The stabilization activity would not extend F-Canyon dissolution operations beyond the time projected for the Mark-31 targets [i.e., about 6 months after the start of stabilization (dissolution activities)]. The time to accomplish the other alternatives would range from about 3-1/2 years to more than 20 years. The accelerated schedule for the alternative that involves dry storage would require more than 5 years to complete all operations, which would be at least 2 years longer than the schedule for the Processing to Metal Alternative.

The Processing to Metal Alternative would produce a material that would be the least proliferation resistant of the alternatives. However, the amount of plutonium-239 contained in the material is very small in comparison to the existing inventory of weapons-usable plutonium metal currently stored at the SRS and would be a fraction of the total DOE inventory. In addition, DOE does not believe the fuel and slugs contain enough plutonium to increase proliferation concerns regardless of the stabilization alternative. DOE has committed to not use plutonium-239 separated as a result of stabilization, phaseout, shutdown, or cleanout of weapons complex facilities for nuclear explosives purposes. L11-1

DOE proposes to stabilize the 81 canisters of TRR fuel and the one canister of EBR-II slugs now for the reasons described above for the Mark-31 targets.

2.8 Other Alternatives Considered

DOE identified several alternatives that it eliminated from detailed study because they increased environmental or other risks without commensurate benefits or because they would be inconsistent with National Environmental Policy Act requirements for interim actions. These include processing to include fission products (to make the material self-protecting), transporting material off the Site, and burial.

DOE considered the addition of fission products to increase the radioactivity of the stabilized form of the material (e.g., metal). Such an addition would make the material essentially "self-protecting" from theft or potential use in weapons because of high radiation levels. However, this method would result in increased exposures to personnel performing processing and handling operations (e.g., at FB-Line). DOE considers such increased exposures to personnel to be unwarranted and, therefore, did not consider this a reasonable alternative.

Offsite transportation and onsite burial could reduce SRS risks but are disposition alternatives that could limit the choices of alternatives in the ongoing *Programmatic Environmental Impact Statement for Storage and Disposition of Weapons-Usable Fissile Materials* (59 FR 31985). This would be contrary to National Environmental Policy Act requirements and, therefore, DOE did not consider this a reasonable alternative.

2.8.1 INTERIM CANNING AND STORAGE OF FUEL AND TARGETS

DOE considered an approach to temporary wet storage of the fuel and targets stored in the reactor disassembly and canyon basins at the SRS similar to that proposed for the degraded fuel stored at the K-Basin at the Hanford Site in Washington. The first step would be to place the fuels, targets, and sludge in wet or damp inerted Multi-Canister Overpacks and transfer the overpacked materials to a Canister Storage Building (CSB). The second step would be to transfer the material in Multi-Canister L11-1 Overpacks to a Conditioning Facility where it would be dried and passivated. The material would then be returned to the CSB in dry, inerted Multi-Canister Overpacks for extended interim storage.

However, the SRS has neither a Canister Storage Building nor a Conditioning Facility. DOE estimates that the times required for design, project authorization, Congressional approval, construction, and operation of a CSB and Conditioning Facility would be similar to the facilities described for the Dry Storage Alternatives for Mark-31 targets and Mark-16 and -22 fuels described in this EIS (see Section 2.3 and Appendix C). The SRS has already placed the corroding Mark-31 targets in stainless-steel boxes in the disassembly basins. The SRS has ongoing projects and programs to upgrade the storage racks in one of the basins and to improve the water chemistry. As described in Chapter 2, the SRS could place the fuel and target materials in cans and maintain their storage in the basin if the corrosion rate accelerates or the radioactivity levels and basin water chemistry cannot be maintained within acceptable levels. Canning of the fuel and targets would be a compensatory action similar to the first step proposed for the fuel at K-Basin described above. Thus, this approach was not analyzed separately in this EIS because of the similarity to ongoing actions at the SRS and the dry storage technology described for the Improving Storage Alternative in this EIS.

2.8.2 POTENTIAL PROCESSING TECHNOLOGIES

Under this alternative DOE would initiate a development program for design and construction of a new facility for processing Mark-16 and -22 fuels stored at SRS. The purpose of the facility would be to change the fuel to a form that is stabilized without necessarily separating the fissile material from fission products and other materials, such as aluminum. There are a number of technologies DOE could consider under such a development program. The potential environmental impacts from the construction and operation of a new facility cannot be estimated at this time because the technologies are still developmental and the facility has not been designed. Implementation of any of these technologies would require additional National Environmental Policy Act review.

A number of these developmental technologies have progressed beyond initial feasibility studies to cost and schedule estimates. Some of the developmental technologies and criticality prevention techniques are described below. These processes have centered on the disposition of aluminum-clad highly enriched uranium fuel and not on interim stabilization techniques. Research on reactor irradiated materials with significant quantities of plutonium-239, such as the Mark-31 targets, is less developed and the processes described below may not be applicable.

1. Chop and Dilute. The fuel would be mechanically chopped into small pieces and pieces of depleted uranium alloy would be added for criticality control. L11-6

2. Chop and Poison. The fuel would be mechanically chopped into small pieces and a neutron poison would be added for criticality control.

3. Melt and Dilute. The fuel would be melted along with depleted uranium to create a low enriched uranium mixture after cooling.

4. Melt and Poison. The fuel would be melted and a neutron poison would be added for criticality control after cooling.

5. Dissolve and Dilute. The fuel would be dissolved and depleted uranium would be added to create a low enriched uranium mixture. No separation or removal of fission products, aluminum, or fissile material would occur. The resulting solution would be vitrified.

6. Dissolve and Poison. The fuel would be dissolved and a neutron poison would be added to create a dilute mixture. No separation or removal of fission products, aluminum, or fissile material would occur. The resulting mixture would be vitrified.

In processes 1, 2, 3 and 4, the fuel would be dry-stored in canisters after processing. The processing facility and the dry storage facility would require an estimated 10 years for construction and operation.

L11-6 In processes 5 and 6, the fuel would be dry-stored until a new dissolution facility could be built adjacent to the Defense Waste Processing Facility. The fuel would be dissolved, mixed with either a neutron poison or depleted uranium, and vitrified in the DWPF. The Dry Storage Facility and the dissolution facility would both require construction. As described in Section 2.3, DOE expects a dry storage facility could take as long as 5 to 10 years for construction and operation. A new dissolution facility would take as much or more time for construction and operation. In addition, as explained in Section 2.3 for the Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative, there is considerable technical uncertainty about using existing DWPF processes and equipment for this purpose.

2.9 Facility Utilization

TC The stabilization and storage of the materials covered by this EIS involve the utilization of various storage and processing facilities at the SRS, as described in this chapter and in Appendix C. For certain materials, such as the Mark-16 and -22 fuel and the plutonium and uranium stored in vaults, the preferred alternative could be achieved through the use of several different groups of facilities; the environmental impacts of these different paths are within the impacts described in Chapters 4 and 5. The facilities utilized for these materials, the sequence of stabilizing the different materials, the potential facility utilization for actions outside the scope of this EIS (such as the Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel EIS and the Programmatic EIS for Storage and Disposition of Weapons-Usable Fissile Materials, expected future budget constraints, and nonproliferation objectives) all would significantly affect the strategy for the operation and ultimate shutdown of these facilities. This overall strategy, in turn, affects the cost and other programmatic attributes of the stabilization activities described in this EIS. Thus, to support the decisionmaking process for this EIS and related NEPA activities, DOE has initiated a study of facility utilization strategies to ensure that these decisions reflect the optimum strategy for the Department.

The key factors in this study to identify optimum strategies include the following:

- Consideration of health and safety risk
- Reduction of life-cycle and annualized cost
- Furtherance of nonproliferation policy goals through early shutdown of reprocessing capability (such as the F-Canyon PUREX process)
- Consistency with the Defense Nuclear Facility Safety Board recommendation for material stabilization

DOE is developing three primary cases:

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- Materials stabilization without the restart of the H-Canyon dissolution and separation processes or the startup of associated HB-Line facilities (this case emphasizes annualized cost reduction)
- Materials stabilization with the restart of H-Canyon and the startup of associated HB-Line facilities and the shutdown of the F-Canyon dissolution and separation processes following potential limited stabilization activities (this case emphasizes nonproliferation policy goals)
- Optimized materials stabilization activities between F- and H-Area facilities (this case is based on equal consideration of all factors)

DOE will complete the study before making its decisions on this EIS to ensure that the decisions are consistent with an integrated facility utilization strategy for the SRS chemical processing and nuclear material storage facilities that achieves the Department's programmatic goals.

CHAPTER 3. AFFECTED ENVIRONMENT

The activities described in this environmental impact statement would take place on the Savannah River Site, primarily in industrialized areas (Figures 2-1 and 2-2). The only exceptions would involve the interarea transport of nuclear materials or waste and the potential construction of a facility that would provide dry storage of spent fuel. The industrialized areas consist primarily of buildings, paved parking lots, and graveled areas. While there are some grassed areas around the administration buildings and there is vegetation along drainage ditches, most of these industrialized areas have little or no vegetation. Therefore, they have minimal value as wildlife habitat. There are no aquatic habitats or wetlands in these areas, nor are there any threatened or endangered species. No SRS facilities have been nominated for inclusion in the National Register of Historic Places, and there are no plans for such nomination. Because the F- and H-Areas are industrial sites constructed during the 1950s, the presence of any important cultural resources is unlikely.

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DOE has identified an undeveloped host site for the potential construction of a Dry Storage Facility. This site is to the south and east of H-Area, adjacent to SRS Road E and close to an existing railroad line (Figure 3-1). DOE could connect this site to existing electricity, water, and steam networks with minimal additional construction.

The host site is representative of many areas on the SRS that could support stabilization activities. It is almost completely forested, for the most part with 5- to 40-year-old upland pine. The Savannah River Forest Station (which is operated by the U.S. Forest Service) conducts an active management program on these forested lands. The site contains suitable habitat for white-tailed deer and feral hogs as well as other species common to the mixed pine/hardwood forest of South Carolina. Sections 4.1 and 4.2 describe impacts associated with the operation of the Dry Storage Facility, and Section 4.3 describes its construction impacts.

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DOE would transport nuclear material or waste using existing SRS roads or railways. The primary SRS roadways (see Figure 3-2) are in good condition, smooth and free from potholes. Railings along the roadways offer protection at appropriate locations from dropoffs or other hazards. In general, traffic is heavy in the early morning and late afternoon when workers commute to and from the Site. Railroads on the Site include both CSX and SRS track lines. The rails and crossties are in good condition, and the track lines are clear of vegetation and debris. The rail lines cross the surface waters, floodplains, and wetlands associated with Upper Three Runs Creek, Fourmile Branch, and Pen

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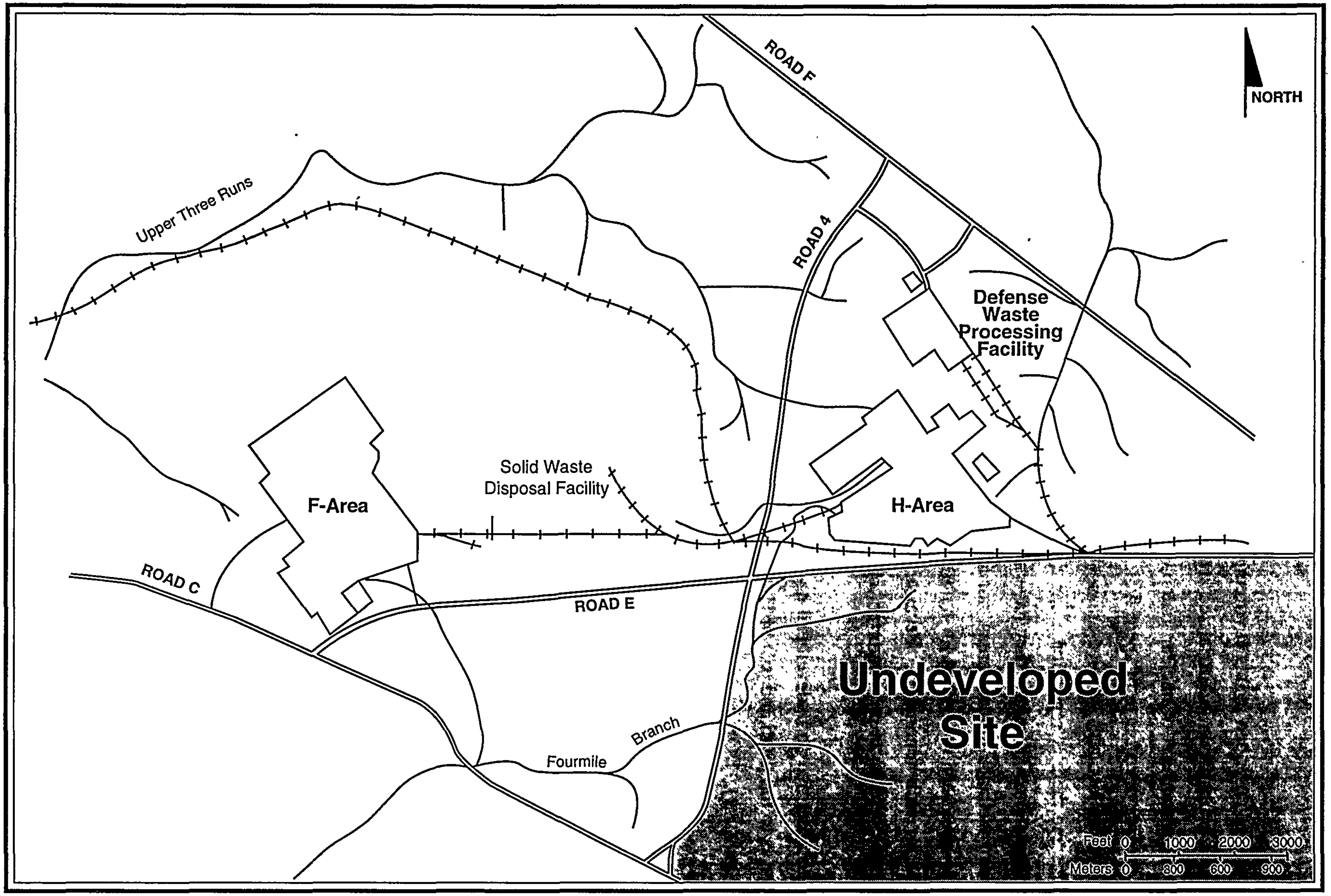


Figure 3-1. Representative site for Dry Storage Facility.

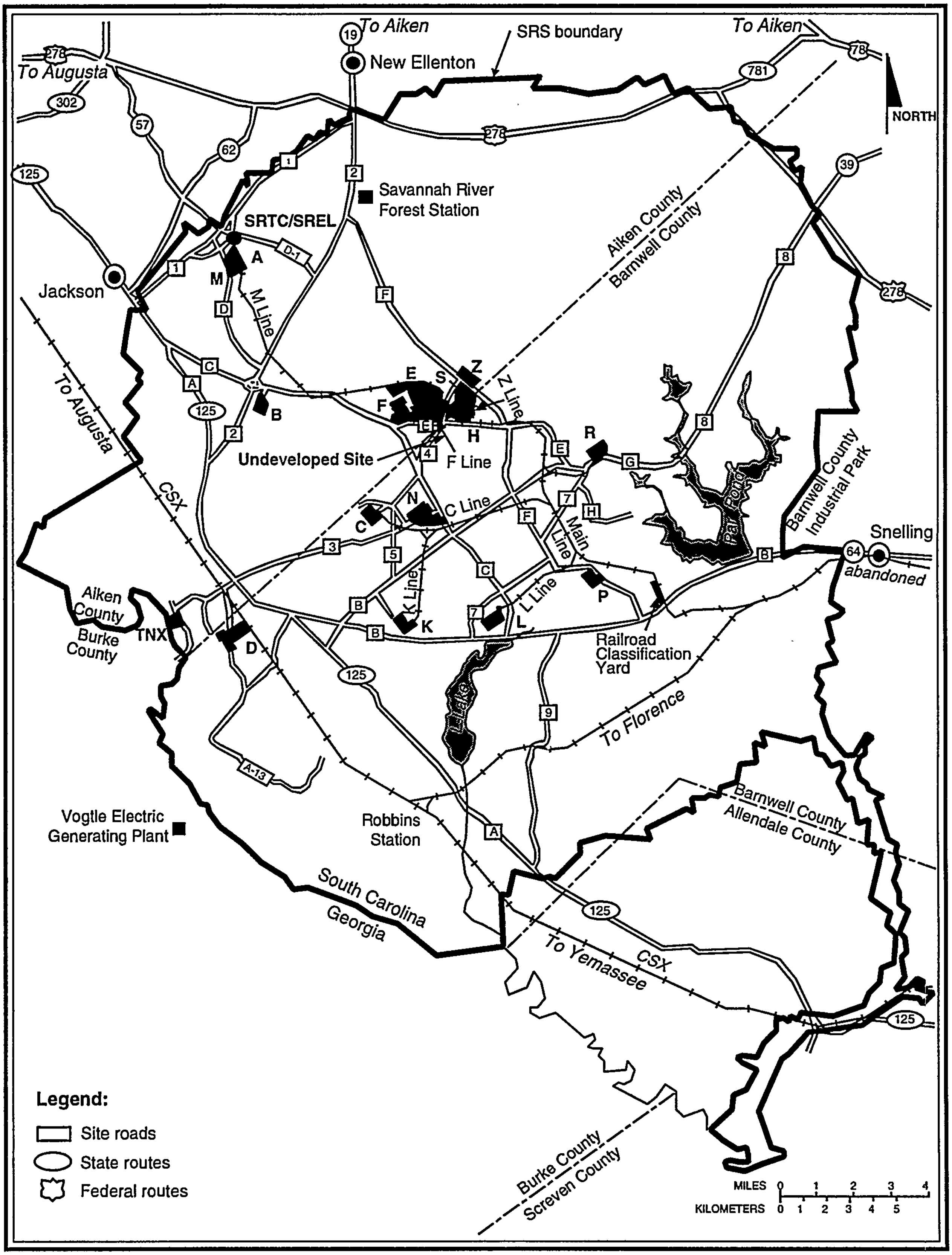


Figure 3-2. Savannah River Site, showing principal industrial areas, roads, and railroads.

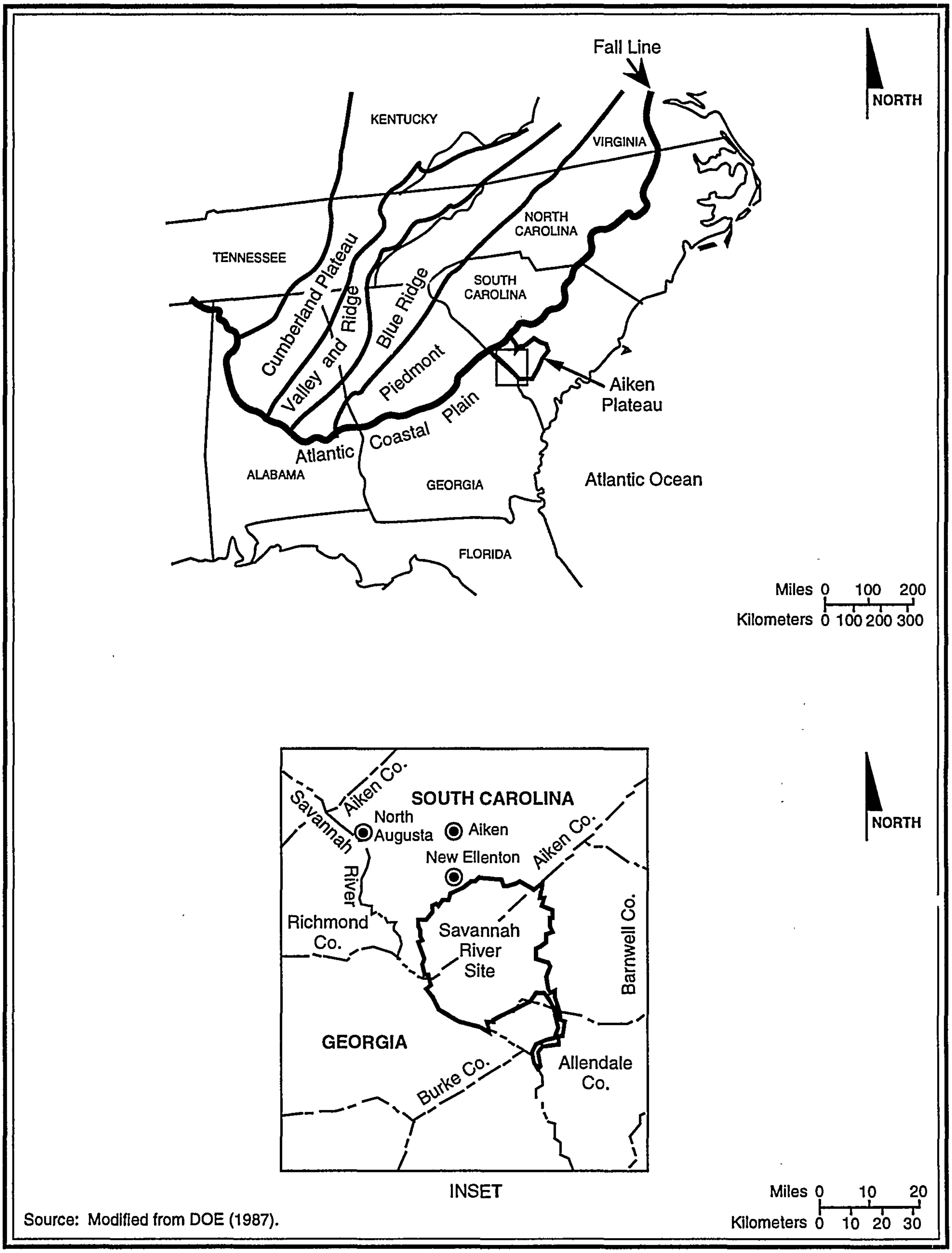
Branch. There is a Carolina Bay along the K-Line railway and an abandoned farm pond near the L-Line railway. A number of documents (Wike et al. 1993; Wiener and Smith 1981; Bennett and TE McFarlane 1983; Gibbons, McCort, and Mayer 1986; Whicker 1988; Workman and McLeod 1990; and Cothran et al. 1991) provide detailed ecological information, including habitat descriptions and animal species lists. Several monographs (Patrick, Cairns, and Roback 1967; Dahlberg and Scott 1971; Bennett and McFarlane 1983), the eight-volume comprehensive cooling water study (Du Pont 1987), and three EISs (DOE 1984, 1987, 1990) that evaluated operations of SRS production reactors describe the aquatic systems and biota of the SRS.

3.1 Geologic Setting and Seismicity

The Savannah River Site is on the Aiken Plateau of the Upper Atlantic Coastal Plain about 40 kilometers (25 miles) southeast of the Fall Line that separates the Atlantic Coastal Plain from the Piedmont (Figure 3-3). Most of the nuclear material storage areas considered in this EIS are on topographically high (upland) areas that are generally flat and lack any distinctive features. The range of local relief of these areas above nearby lowlands is from 12 meters (40 feet) in L-Area to about 60 meters (190 feet) in F-Area. Local relief above nearby lowlands reaches about 55 meters (180 feet) in M-Area, 50 meters (160 feet) in H-Area, 30 meters (90 feet) in K-Area, and 25 meters (80 feet) in P-Area. All storage areas are above the 100-year floodplain. The *Final Environmental Impact Statement, Continued Operation of K-, L-, and P-Reactors, Savannah River Site, Aiken, South Carolina* (DOE 1990) contains a complete description of the geologic setting and the stratigraphic sequences of the SRS. The *Soil Survey of Savannah River Plant Area, Parts of Aiken, Barnwell, and Allendale Counties, South Carolina* (USDA 1990) describes soil characteristics and erosion potential for the area.

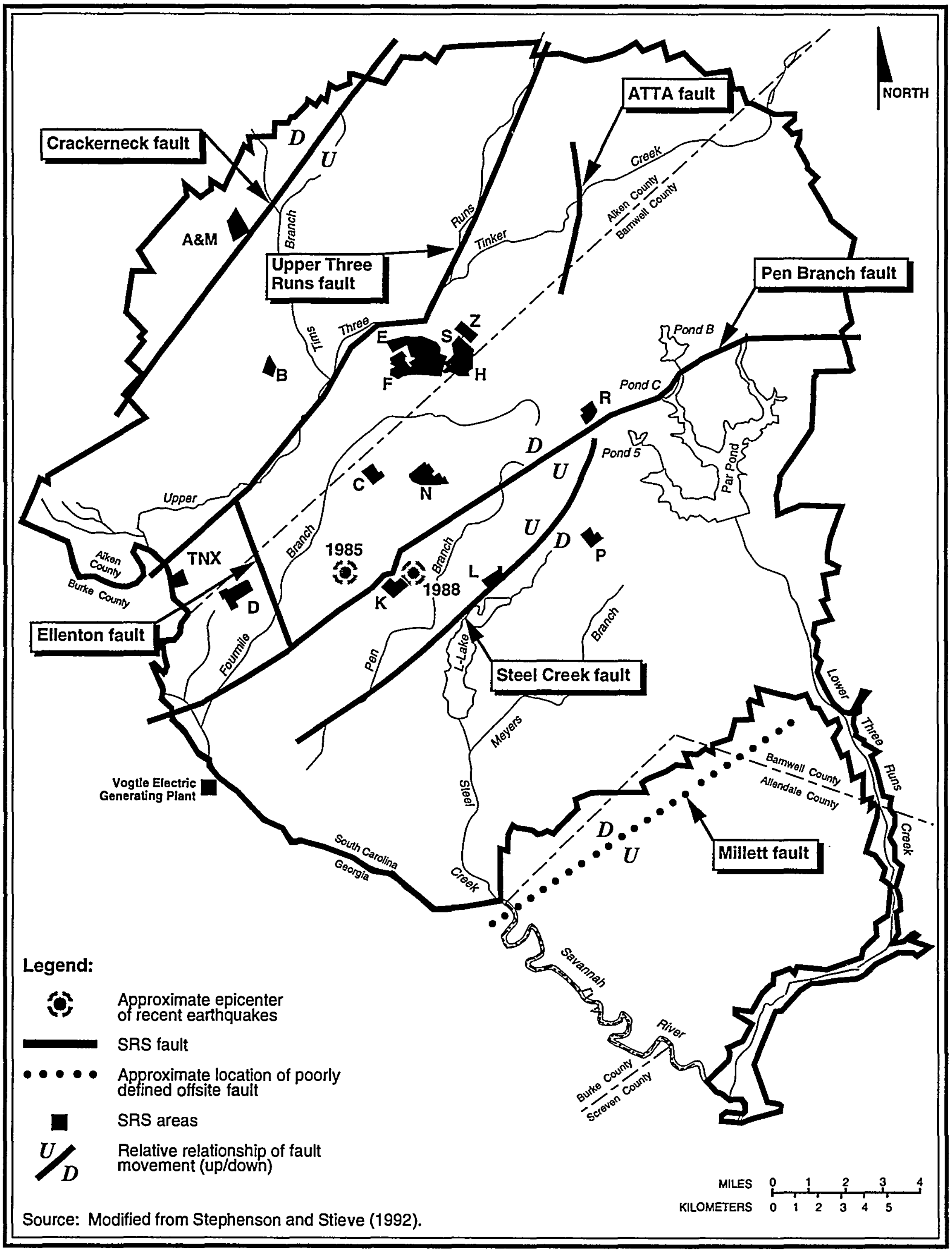
3.1.1 SUBSURFACE FEATURES

TE There are several fault systems off the Site northwest of the Fall Line (DOE 1990). A recent study of available geophysical evidence (Stephenson and Stieve 1992) identified six faults under the SRS: the Pen Branch, Steel Creek, Advanced Tactical Training Area (ATTA), Crackerneck, Ellenton, and Upper Three Runs Faults. Figure 3-4 shows the locations of these faults. The closest of these to areas storing nuclear materials are the Steel Creek Fault, which passes through L-Area, and the Pen Branch Fault, which passes close to K-Area. The fault lines on Figure 3-4 represent the projection of the faults to the ground surface; the actual faults do not reach the surface but stop several hundred feet below it. Based on information developed to date, none of the faults discussed in this section is "capable." A fault is capable if it has moved at or near the ground surface within the past



Source: Modified from DOE (1987).

Figure 3-3. General location of the Savannah River Site and relationship to physiographic provinces of the eastern United States.



Source: Modified from Stephenson and Stieve (1992).

Figure 3-4. Savannah River Site, showing seismic fault lines and locations of onsite earthquakes.

35,000 years or is associated with another fault that has moved in the past 35,000 years. (For a more detailed definition of a capable fault, see 10 CFR Part 100.)

3.1.2 SEISMICITY

Two major earthquakes have occurred within 300 kilometers (186 miles) of the SRS. The first was the Charleston, South Carolina, earthquake of 1886, which had an estimated Richter scale magnitude of 6.8 and occurred approximately 145 kilometers (90 miles) from the Site. The SRS area experienced an estimated peak horizontal acceleration of 10 percent of gravity (0.10*g*) during this earthquake (URS/Blume 1982). The second major earthquake was the Union County, South Carolina, earthquake of 1913, which had an estimated Richter scale magnitude of 6.0 and occurred about 160 kilometers (99 miles) from the Site (Bollinger 1973). Because these earthquakes are not associated conclusively with a specific fault, researchers cannot determine the amount of displacement resulting from them.

In recent years, two earthquakes occurred inside the SRS boundary. On June 8, 1985, an earthquake with a local Richter scale magnitude of 2.6 and a focal depth of 0.96 kilometer (0.59 mile) occurred on the Site; its epicenter was west of C- and K-Areas. On August 5, 1988, an earthquake with a local Richter scale magnitude of 2.0 and a focal depth of 2.68 kilometers (1.66 miles) occurred on the Site; its epicenter was northeast of K-Area. Existing information does not correlate the two earthquakes conclusively with the known faults on the Site. Figure 3-4 shows the locations of the epicenters of these two earthquakes.

Outside the SRS boundary, a Richter scale magnitude 3.2 earthquake occurred on August 8, 1993, approximately 16 kilometers (10 miles) east of the City of Aiken near Couchton, South Carolina. Residents reported feeling this earthquake in Aiken, New Ellenton (immediately north of the SRS), and North Augusta [approximately 40 kilometers (25 miles) northwest of the SRS], and on the Site.

The accident analyses for this EIS evaluated a severe earthquake of a magnitude that would produce a peak ground acceleration of 0.2*g*, which is estimated to recur at an interval of about once every 5,000 years. The EIS analyzes earthquakes of this magnitude because this represents the SRS design-basis earthquake (i.e., new facilities would be designed to withstand an earthquake of this magnitude). The canyon structures were designed to resist a bomb blast impact against the exterior walls. The acceleration of the blast "front" from a nearby detonation would be many times the acceleration due to gravity (32 feet per second squared). For this reason, the structures would be highly damage-resistant to an earthquake with a horizontal ground acceleration of 0.20*g* or 20 percent of gravity at

TC the structure base, although some materials probably would be released. A precise translation of this acceleration to a Richter scale reading is not possible because the impact would be greatly affected by the type of soil in the area of the earthquake epicenter, the nearness of a shallow fault line, and attenuation of the shock wave in rock or other formations.

3.2 Water Resources

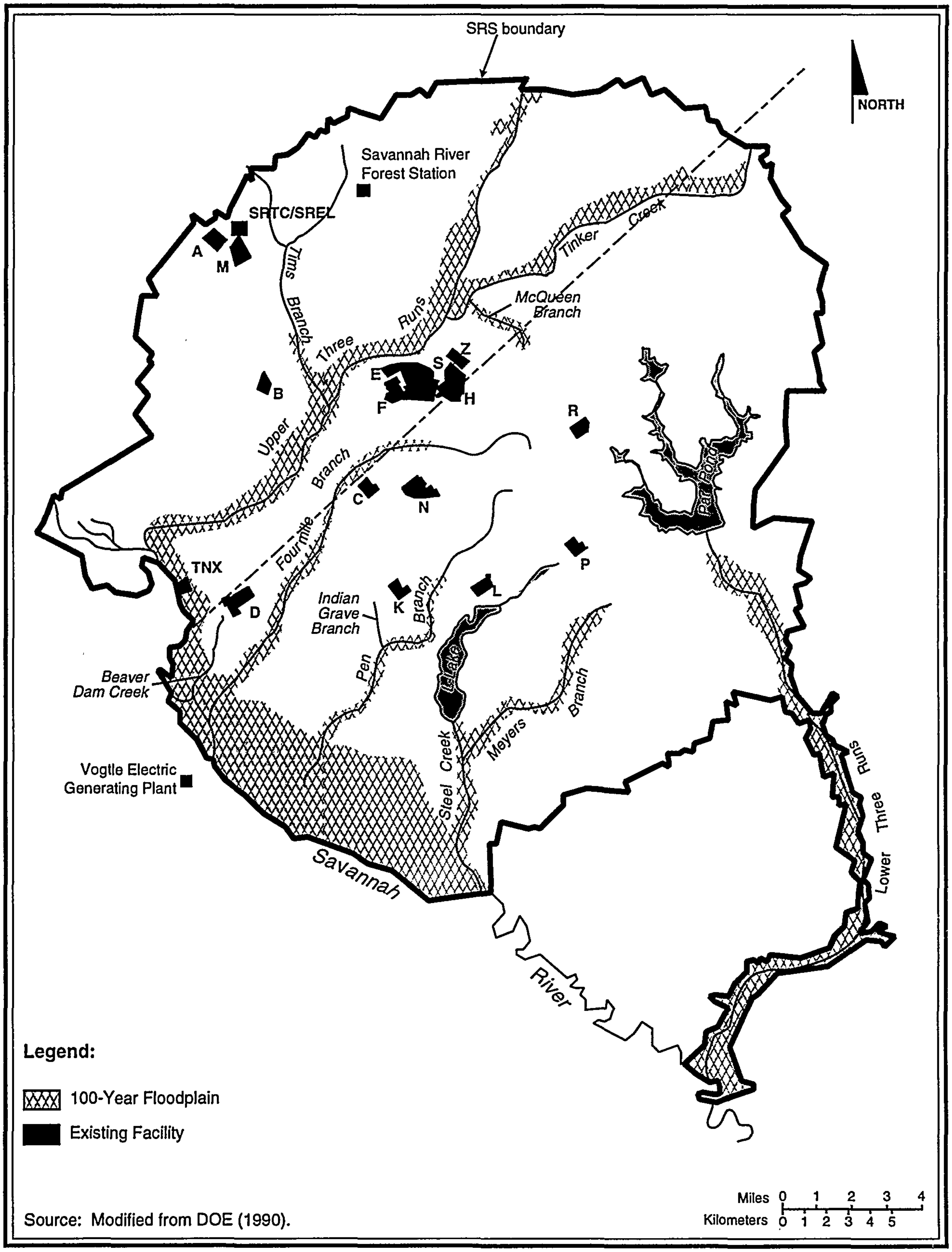
3.2.1 SURFACE-WATER AND GROUNDWATER FEATURES

Six tributaries of the Savannah River - Upper Three Runs Creek, Fourmile Branch, Beaver Dam Creek, Pen Branch, Steel Creek, and Lower Three Runs Creek - drain almost all of the SRS (Figure 3-5). Surface waters in the vicinity of the F- and H-Areas flow into Upper Three Runs Creek and Fourmile Branch. Similarly, shallow groundwater in the vicinity recharges both Upper Three Runs Creek and Fourmile Branch.

TE The Savannah River, which forms the boundary between Georgia and South Carolina, supplies potable water to several municipalities. Upstream from the SRS, the river supplies domestic and industrial water needs for Augusta, Georgia, and North Augusta, South Carolina. Approximately 203 river kilometers (126 river miles) downstream from the SRS, the river supplies domestic and industrial water needs for the Cherokee Hill Water Treatment Plant at Port Wentworth, Georgia, through intakes at river kilometer 47 (river mile 29), and for Beaufort and Jasper Counties in South Carolina through intakes at about river kilometer 63 (river mile 39.2).

Groundwater is a domestic, municipal, and industrial water source throughout the Upper Coastal Plain. Most municipal and industrial water supplies in Aiken County are from the deep aquifers. Domestic water supplies are primarily from the intermediate and shallow zone. In Barnwell and Allendale Counties, the intermediate zone and overlying units that thicken to the southeast supply some municipal users. At the SRS, most groundwater production is from the deep zone, with a few lower capacity wells pumping from the intermediate zone. Every major operating area at the SRS has groundwater wells; total groundwater production is from 34,000 to 45,000 cubic meters (9 to 12 million gallons) per day, similar to the volume pumped for industrial and municipal production within 16 kilometers (10 miles) of the Site (Arnett, Karapatakis, and Mamatey 1993).

Groundwater beneath the Site flows slowly toward SRS streams and swamps and into the Savannah River at rates ranging from inches to several hundred feet per year. The depth to which the onsite



Source: Modified from DOE (1990).

Figure 3-5. Savannah River Site, showing 100-year floodplain and major stream systems.

streams cut into the soils controls the horizontal movement of groundwater. The valleys of the smaller perennial streams allow discharge from the shallow saturated geologic formations. The valleys of major tributaries of the Savannah River (e.g., Upper Three Runs Creek) drain formations of intermediate depth, and the valley of the Savannah River drains deep formations.

Groundwater flow is downward at some locations on the site, including A-, M-, L-, and P-Areas. In other areas, gradient and subsequent water pressure is upward from the lower to the upper sediments. This upward flow occurs, for example, in certain sections of F- and H-Areas and around K-Area. Horizontal groundwater flow occurs at the M-Area metallurgical laboratory (to the west-northwest in the shallow aquifer and subsequent flow to the south toward Upper Three Runs Creek in the intermediate aquifer), K-Area disassembly basin (toward Pen Branch and L-Lake), P-Area disassembly basin (toward Steel Creek), F-Canyon building (toward Upper Three Runs Creek and Fourmile Branch), and H-Canyon building (toward Upper Three Runs Creek and its tributaries).

3.2.2 SURFACE-WATER AND GROUNDWATER QUALITY

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In 1993, the major releases of radionuclides to surface waters were 12,700 curies of tritium, 0.477 curie of strontium-89 and -90, and 0.246 curie of cesium-137. Table 3-1 lists radioactive liquid releases by source for 1993. The resulting doses to a downriver consumer of river water from all radionuclides released from the Site were less than 2 percent of the EPA and DOE standards for public water supplies (40 CFR Part 141 and DOE Order 5400.5, respectively) and less than 0.2 percent of the DOE dose standard from all pathways (DOE 5400.5). From a nonradiological perspective, there was no significant difference between upriver and downriver water quality parameters. Other than 72 instances of exceeding coliform standards (an indicator of the presence of human or animal fecal material), analyses of streams, including the Savannah River, that can receive SRS discharges met the more stringent 1992 updated river classification of Freshwaters; that is, 99.9 percent of the analyses were in compliance with the SRS National Pollutant Discharge Elimination System permit.

TE

Industrial solvents, metals, tritium, and other constituents used or generated on the Site have contaminated the shallow aquifers beneath 5 to 10 percent of the SRS. These aquifers are not used for SRS operations or drinking water; however, they do discharge to Site streams and eventually the Savannah River. Figure 3-6 shows groundwater contamination on the Site (Arnett, Karapatakis, and Mamatey 1993). Most contaminated groundwater at the SRS flows beneath a few facilities; contaminants reflect the operations and chemical processes performed at those facilities. At F- and H-Areas, contaminants in the groundwater include tritium and other radionuclides, metals, nitrates,

Table 3-1. 1993 liquid releases by source (including direct and seepage basin migration releases).[a]

		Curies					
Radionuclide[b]	Half-life (years)	Reactors	Separations[c]	Reactor materials	Heavy water	SRTC/ TNX	Total
H-3 (oxide)	12.3	2,290	9,880		499	0.129	12,700
Sr-89,90[d]	29.1	0.187	0.241		4.65×10^{-2}	2.02×10^{-3}	0.477
I-129	1.6×10^{7}		2.20×10^{-2}				2.20×10^{-2}
Cs-137	30.2	1.29×10^{-2}	0.233				0.246
Pm-147	2.6		7.03×10^{-3}				7.03×10^{-3}
U-235,238	4.5×10^{9}		1.14×10^{-5}				1.14×10^{-5}
Pu-239[e]	24,000	5.97×10^{-4}	8.65×10^{-3}	7.64×10^{-5}		2.66×10^{-4}	9.59×10^{-3}

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a. Source: Arnett, Karapatakis, and Mamatey (1994).
b. H = hydrogen (H-3 = tritium), Sr = strontium, I = iodine, Cs = cesium, Pm = promethium, U = uranium Pu = plutonium.
c. Includes F- and H-Area releases.
d. Includes unidentified beta-gamma.
e. Includes unidentified alpha.

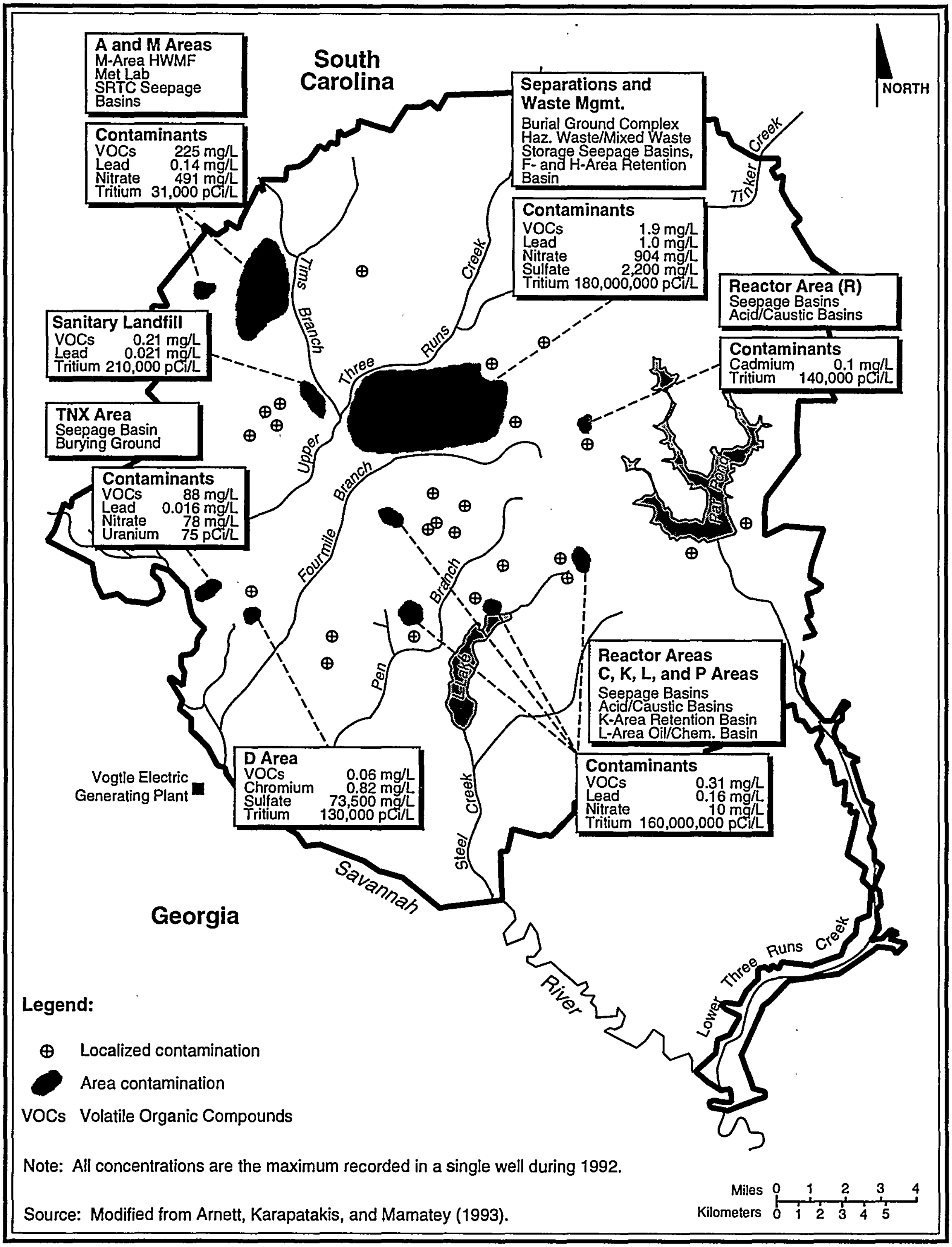
and chlorinated and volatile organics. At A- and M-Areas, contamination includes chlorinated volatile organics, radionuclides, metals, and nitrates. At the reactors (K-, L-, and P-Areas), tritium, other radionuclides, and lead are in the groundwater.

Radioactive constituents (tritium, cesium-137, iodine-131, ruthenium-106, and strontium-89 and -90) above drinking water standards have occurred in F-Area monitoring wells. One well (FCA-9DR) showed activities considerably higher than others; strontium activities were especially notable, as much as 1,000 times over drinking water standards (Arnett, Karapatakis, and Mamatey 1994). Studies of flow directions, infiltration rates, and operating history indicate that this contamination is from an isolated incident that occurred more than 35 years ago (Reed 1993).

Contamination beneath the H-Canyon reflects the pervasiveness of tritium in the H-Area. The tritium is not directly from H-Canyon activities, but rather results from past use of the nearby H-Area seepage basins with subsequent transport beneath the canyon.

3.3 Air Resources

Based on SRS data collected from onsite meteorological towers for the 5-year period 1987 through 1991, maximum wind direction frequencies are from the northeast and west-southwest and the average wind speed is 3.8 meters per second (8.5 miles per hour) (Shedrow 1993). The average annual temperature at the SRS is 17.8°C (64°F). The atmosphere in the SRS region is unstable approximately 56 percent of the time, neutral 23 percent of the time, and stable about 21 percent of the time (Shedrow 1993). In general, as the atmosphere becomes more unstable, atmospheric dispersion of airborne pollutants increases and ground-level pollutant concentrations decrease. TE



Note: All concentrations are the maximum recorded in a single well during 1992.

Source: Modified from Arnett, Karapatakis, and Mamatey (1993).

Figure 3-6. Groundwater contamination at the Savannah River Site.

3.3.1 SEVERE WEATHER CONDITIONS

The SRS area experiences an average of 55 thunderstorm days per year with 50 percent of these occurring in June, July, and August (Shedrow 1993). On an annual average, lightning flashes will strike six times per year on a square-kilometer area (Hunter 1990). The highest windspeed recorded at Bush Field (Augusta, Georgia) between 1950 and 1990 was 100 kilometers (62 miles) per hour (NOAA 1990).

From 1954 to 1983, 37 reported tornadoes occurred in a 1-degree square of latitude and longitude that includes the SRS (WSRC 1993a). This frequency of occurrence is equivalent to an average of about one tornado per year. The estimated probability of a tornado striking a point on the SRS is 0.0000711 per year. This results in a "point-strike recurrence" interval of about once every 14,000 years (Bauer et al. 1989). Due to the size of the SRS, the occurrence of several individual strikes is unlikely. Since operations began at the SRS in 1953, nine tornadoes have been confirmed on or near the Site. Nothing more than light damage was reported, with the exception of a tornado in October 1989 that caused considerable damage to forest resources in an undeveloped southeastern sector of the SRS (Shedrow 1993).

From 1700 to 1992, 36 hurricanes occurred in South Carolina, resulting in an average frequency of about one hurricane every 8 years (WSRC 1993a). Because SRS is about 160 kilometers (100 miles) inland, the winds associated with hurricanes have usually diminished below hurricane force [i.e., equal to or greater than a sustained wind speed of 33.5 meters per second (75 miles per hour)] before reaching the SRS. Winds exceeding hurricane force have been observed only once at the SRS (Hurricane Gracie in 1959) (Shedrow 1993).

3.3.2 RADIOLOGICAL AIR QUALITY

DOE provides detailed summaries of radiological releases to the atmosphere from SRS operations, along with the resulting concentrations and doses, in a series of annual environmental data reports. This section references several of these documents, which contain additional information. The information enables comparisons of current data with releases, concentrations, and doses associated with each alternative.

In the SRS region, airborne radionuclides originate from natural sources (i.e., terrestrial and cosmic), worldwide fallout, and Site operations. The SRS maintains a network of air monitoring stations on and around the Site to determine concentrations of radioactive particulates and aerosols in the air (Arnett, Karapatakis, and Mamatey 1994).

Table 3-2 lists average and maximum nontritium atmospheric radionuclide concentrations at the SRS boundary and at background monitoring locations [160-kilometer (100-mile) radius] during 1993. Tritium is the only radionuclide of SRS origin detected routinely in offsite air samples above background (control) concentrations (Cummins, Martin, and Todd 1990, 1991; Arnett et al. 1992; Arnett, Karapatakis, and Mamatey 1993). Table 3-3 lists average concentrations of tritium in the atmosphere, as measured at the boundary and offsite monitoring locations.

Table 3-2. Radioactivity in air at the SRS boundary and at the 160-kilometer (100-mile) radius during 1993 (picocuries per cubic meter).[a]

Location	Gross alpha	Nonvolatile beta	Sr-89,90[b]	Pu-238[b]	Pu-239[b]
Site boundary					
Average	0.0018	0.019	<0.000088	0.00000052	0.00000026
Maximum	0.0050	0.063	0.00027	0.0000048	0.0000021
Background (160-kilometer radius)					
Average	0.0020	0.020	<0.00027	0.00000070	<0.0000020
Maximum	0.0049	0.043	0.00058	0.0000059	0.0000044

a. Source: Arnett (1994).
b. Monthly composite.

Table 3-3. Average atmospheric tritium concentrations around the Savannah River Site (picocuries per cubic meter).[a]

Location	1993	1992	1991
Site boundary	30	27	21
40-kilometer radius	9	11	11
160-kilometer radius	4.7	8.3	8.5

a. Source: Arnett, Karapatakis, and Mamatey (1994).

Table 3-4 lists 1993 radionuclide releases from each major operational group of SRS facilities. All radiological impacts are within regulatory requirements.

3.3.3 NONRADIOLOGICAL AIR QUALITY

TE The SRS is in the Augusta (Georgia) - Aiken (South Carolina) Interstate Air Quality Control Region. This region, which is designated as a Class II area, is in compliance with National Ambient Air Quality Standards (NAAQS) for criteria pollutants. Class II is the initial designation of any area that is not considered a pristine area; pristine areas include national parks or national wilderness areas. The

Table 3-4. 1993 atmospheric releases by operational group.[a]

Radionuclide	Half-life	Reactors	Separations[b]	Reactor materials	Heavy water	SRTC[c]	Diffuse and fugitive[d]	Total
		Operational group (curies)						
GASES AND VAPORS								
H-3 (oxide)	12.3 yrs	38,500	93,900		448		43.1	133,000
H-3 (elem)	12.3 yrs		58,200					58,200
H-3 Total	12.3 yrs	38,500	152,000		448		43.1	191,000
C-14	5,700		1.69×10^{-2}				4.00×10^{-6}	1.69×10^{-2}
I-129	1.6×10^{7} yrs		4.96×10^{-3}				6.88×10^{-7}	4.96×10^{-3}
I-131	8 days		8.89×10^{-5}			5.92×10^{-5}		1.48×10^{-4}
I-133	20.8 hrs					1.96×10^{-3}		1.96×10^{-3}
Xe-135	9.1 hrs					3.19×10^{-2}		3.19×10^{-2}
PARTICULATES								
Ni-63	100 yrs						2.00×10^{-7}	2.00×10^{-7}
Co-60	5.3 yrs		5.89×10^{-9}				3.34×10^{-17}	5.89×10^{-9}
S-35	87.2 days						2.00×10^{-6}	2.00×10^{-6}
Sr-89,90[e]	29.1 yrs	1.81×10^{-4}	1.88×10^{-3}	8.32×10^{-5}	7.19×10^{-6}	1.19×10^{-5}	1.11×10^{-4}	2.27×10^{-3}
Zr-95	64 days						2.39×10^{-14}	2.39×10^{-14}
Ru-106	1.0 yrs	3.99×10^{-6}	5.76×10^{-9}				4.96×10^{-12}	4.00×10^{-6}
Sb-125	2.8 yrs						7.27×10^{-15}	7.27×10^{-15}
Cs-134	2.1 yrs						1.40×10^{-17}	1.49×10^{-6}
Cs-137	30.2 yrs	1.04×10^{-4}	1.49×10^{-6}			1.51×10^{-6}	4.33×10^{-11}	6.34×10^{-4}
Ce-144	285 days						1.13×10^{-13}	1.13×10^{-13}
Eu-154	8.6 yrs						3.44×10^{-13}	3.44×10^{-13}
Eu-155	4.7 yrs						1.63×10^{-13}	1.63×10^{-13}
U-235,238	4.5×10^{9} yrs		1.86×10^{-3}	1.55×10^{-5}		2.89×10^{-8}	4.74×10^{-5}	1.92×10^{-3}
Pu-238	87.7 yrs		1.21×10^{-3}			1.00×10^{-8}	4.63×10^{-12}	1.21×10^{-3}
Pu-239[f]	2.4×10^{4} yrs	4.11×10^{-6}	1.06×10^{-3}	3.50×10^{-6}	8.42×10^{-7}	9.41×10^{-6}	4.70×10^{-7}	1.08×10^{-3}
Am-241,243	7.4×10^{3} yrs		1.42×10^{-4}			1.34×10^{-6}	8.86×10^{-13}	1.43×10^{-4}
Cm-242,244	18.1 yrs		4.96×10^{-5}			6.83×10^{-6}	7.33×10^{-12}	5.64×10^{-5}

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a. Source: Arnett (1994).
b. Includes both F- and H-Area releases.
c. SRTC - Savannah River Technology Center.
d. Estimated releases from minor unmonitored diffuse and fugitive sources.
e. Includes unidentified beta-gamma emissions.
f. Includes unidentified alpha emissions.

criteria pollutants include sulfur dioxide, nitrogen oxides (reported as nitrogen dioxide), particulate matter (less than or equal to 10 microns in diameter), carbon monoxide, ozone, and lead (40 CFR Part 50).

DOE utilized the comprehensive emissions inventory data for 1990, which is the most recent data available, to establish the baseline year for showing compliance with national and state air quality standards by calculating actual emission rates for existing sources. DOE based its calculated emission rates for the sources on process knowledge, source testing, material balance, and U.S. Environmental Protection Agency (EPA) Air Pollution Emission Factors (AP-42; EPA 1985). The inventory also included maximum potential emissions for sources permitted for construction through 1992.

DOE has performed atmospheric dispersion modeling for criteria and toxic air pollutants for actual TE | emissions for the baseline year 1990 (plus potential emissions for sources permitted for construction), using the EPA Industrial Source Complex Short Term No. 2 Model. This model used data from the SRS meteorological tower for 1991 along with the 1990 emissions data to estimate maximum ground-level air pollutant concentrations at the SRS boundary. DOE added the incremental impacts associated with the alternatives evaluated in this EIS to the baseline concentrations to estimate total air quality impacts.

The South Carolina Department of Health and Environmental Control (SCDHEC) has air quality regulatory authority over the SRS and determines ambient air quality compliance based on air pollutant emissions and estimates of concentrations at the Site boundary based on atmospheric dispersion modeling. The SRS is in compliance with National Ambient Air Quality Standards for criteria pollutants and gaseous fluoride and with total suspended particulate standards, as required by SCDHEC Regulation R.61-62.5, Standard 2, "Ambient Air Quality Standards" (AAQS). Table 3-5 TE | lists these standards and the results of the atmospheric dispersion modeling for baseline year 1990.

The SRS is in compliance with SCDHEC Regulation R.61-62.5, Standard 8, "Toxic Air Pollutants," TE | which regulates the emission of 257 toxic air pollutants (WSRC 1994b). DOE has identified emission sources for 139 of the 257 regulated air toxics; the modeled results indicate that the Site is in compliance with SCDHEC air quality standards. Table 3-6 lists toxic air pollutants that are the same as those that the alternative actions described in this EIS would emit. Table 3-6 also compares TE | maximum downwind concentrations at the Site boundary for baseline year 1990 to SCDHEC standards for toxic air pollutants.

Table 3-5. Estimated ambient concentrations of criteria air pollutants from SRS sources.[a,b]

Pollutant[c]	Averaging time	Concentration (μg/m^3)[d]	Most stringent AAQS[e] (national or state) (μg/m^3)	Concentration as a percent of AAQS[f] (%)
SO$_2$	Annual	10	80[g]	12.5
	24 hours	185	365[g,h]	50.7
	3 hours	634	1,300[g,h]	48.8
NO$_x$	Annual	4	100[g]	4.0
CO	8 hours	23	10,000[g,h]	0.2
	1 hour	180	40,000[g,h]	0.5
Gaseous fluorides (as HF)	12 hours	0.62	3.7[f]	16.8
	24 hours	0.31	2.9[f]	10.7
	1 week	0.15	1.6[f]	9.4
	1 month	0.03	0.8[f]	3.8
PM$_{10}$	Annual	3	50[g]	6.0
	24 hours	56	150[g]	0.4
O$_3$	1 hour	NA[i]	235[g,h]	NA
TSP	Annual geometric mean	11	75[f]	14.7
Lead	Calendar quarter mean	0.0003	1.5[e]	0.02

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a. Source: WSRC (1994b). TE

b. The concentrations are the maximum values at the SRS boundary.

c. SO$_2$ = sulfur dioxide; NO$_x$ = nitrogen oxides; CO = carbon monoxide; PM$_{10}$ = particulate matter $\leq$ 10μm in diameter; TSP = Total Suspended Particulates, O$_3$ = Ozone.

d. Based on actual emissions from all existing SRS sources plus maximum potential emissions for sources permitted for construction through December 1992.

e. AAQS = Ambient Air Quality Standard.

f. Source: SCDHEC (1976).

g. Source: 40 CFR Part 50.

h. Concentration not to be exceeded more than once a year.

i. NA = Not available.

Table 3-6. Estimated 24-hour average ambient concentrations at the SRS boundary - toxic air pollutants regulated by South Carolina from SRS sources.[a]

Pollutant[b]	Concentration (μg/m^3)[c]	Regulatory standard (μg/m^3)	Concentration as a percent of standard (%)
Benzene	31	150	20.70
Hexane	0.07	200	0.04
Nitric acid	6.70	125	5.40
Sodium hydroxide	0.01	20	0.05
Toluene	1.60	2,000	0.08
Xylene	3.80	4,350	0.09

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a. Source: WSRC (1994b). TE

b. Pollutants listed include air toxics of interest in relation to interim management of nuclear materials alternatives. (Section 5.2 addresses the effects of all air toxics.)

c. Based on actual emissions from existing SRS sources plus maximum potential emissions for sources permitted for construction through December 1992.

3.4 Socioeconomics

This section discusses baseline socioeconomic conditions in a region of influence where approximately 90 percent of the SRS workforce lived in 1992. The SRS region of economic influence includes Aiken, Allendale, Bamberg, and Barnwell Counties in South Carolina, and Columbia and Richmond Counties in Georgia. *Socioeconomic Characteristics of Selected Counties and Communities Adjacent to the Savannah River Site* (HNUS 1992) contains additional information on the economic and demographic characteristics of the six-county region.

3.4.1 EMPLOYMENT

Between 1980 and 1990, total employment in the six-county region increased from 139,504 to 199,161, an average annual growth rate of approximately 5 percent. The unemployment rates for 1980 and 1990 were 7.3 percent and 4.7 percent, respectively (HNUS 1992). In 1995, regional employment will be approximately 242,000. Over the 10-year planning period, employment in the region will increase at a projected average rate of 1 percent per year, reaching approximately 264,000 by 2004 (HNUS 1994a). TE

In Fiscal Year 1992, employment at the SRS was 23,351, approximately 10 percent of regional employment, with an associated payroll of more than $1.1 billion. Due to planned budget reductions, Site employment could decline by as many as 4,200 jobs between 1995 and 1996 (Fiori 1995).

3.4.2 POPULATION

Between 1980 and 1990, the population in the region of influence increased 13 percent, from 376,058 to 425,607. More than 88 percent of the 1990 population lived in Aiken County (28.4 percent), Columbia County (15.5 percent), or Richmond County (44.6 percent). In 1995, the population in the six-county region will be approximately 462,000. Over the 10-year planning period, the regional population will grow at a projected rate of 0.4 percent per year, reaching approximately 479,000 by 2004 (HNUS 1994a). TE According to census data, in 1990 the estimated average number of persons per household in the six-county region was 2.72, and the median age of the population was 31.2 years (HNUS 1992).

3.4.3 COMMUNITY CHARACTERISTICS

Executive Order 12898, *Federal Actions to Address Environmental Justice in Minority Populations and Low-Income Populations*, directs Federal agencies to identify and address, as appropriate, disproportionately high and adverse human health or environmental effects of their programs, policies, and activities on minority and low-income populations. Executive Order 12898 also directs the Administrator of the Environmental Protection Agency to convene an interagency Federal Working Group on Environmental Justice. The Working Group is directed to provide guidance to Federal agencies on criteria for identifying disproportionately high and adverse human health or environmental effects on minority and low-income populations; minority populations are referred to in this EIS as people of color. The Working Group has not yet issued the guidance directed by Executive Order 12898, although it has developed working draft definitions. The definitions used in this analysis are based on the draft working definitions. Further, in coordination with the Working Group, DOE is in the process of developing internal guidance on implementing the Executive Order. TC

The potential offsite health impacts would result from releases to the air and to Savannah River water downstream of the SRS. For air releases, standard population dose analyses are based on an 80-kilometer (50-mile) radius because reasonably foreseeable dose levels beyond that distance would be negligible. For liquid releases, the region of interest includes areas along the river that draw drinking water from the river (Beaufort and Jasper Counties in South Carolina and Port Wentworth in Georgia). Combining these two areas, the analysis included data (U.S. Bureau of the Census 1990a,b) for populations in all census tracts that have at least 20 percent of their area in the 80-kilometer (50-mile) radius and all tracts from Beaufort and Jasper Counties in South Carolina and Effingham and Chatham Counties in Georgia, which are downstream of the Site. DOE used data from each census tract in this combined region to identify the racial composition of communities and the number of persons characterized by the U.S. Bureau of the Census as living in poverty. The combined region contains 247 census tracts, 99 in South Carolina and 148 in Georgia. TC

Tables 3-7 and 3-8 list racial and economic characteristics, respectively, of the population in the combined region. Table 3-7 indicates a total population of more than 993,000 in the area; of that population, approximately 618,000 (62.2 percent) are white. Within the population of people of color, approximately 94 percent are African American. The remainder of the population of people of color is made up of small percentages of Asian, Hispanic, and Native American persons. Figure 3-7 shows the distribution of people of color by census tract areas in the SRS region.

TC **Table 3-7.** General racial characteristics of population in the SRS region of interest.[a]

State	Total population	White	African American	Hispanic	Asian	Native American	Other	People of color	Percent people of color[b]
South Carolina	418,685	267,639	144,147	3,899	1,734	911	355	151,046	36.08%
Georgia	574,982	350,233	208,017	7,245	7,463	1,546	478	224,749	39.09%
Total	993,667	617,872	352,164	11,144	9,197	2,457	833	375,795	37.82%

a. Source: U.S. Bureau of the Census (1990a).

b. People of color population divided by total population.

TC **Table 3-8.** General poverty characteristics of population in the SRS region of interest.[a]

Area	Total population	Persons living in poverty[b]	Percent living in poverty
South Carolina	418,685	72,345	17.28%
Georgia	574,982	96,672	16.81%
Total	993,667	169,017	17.01%

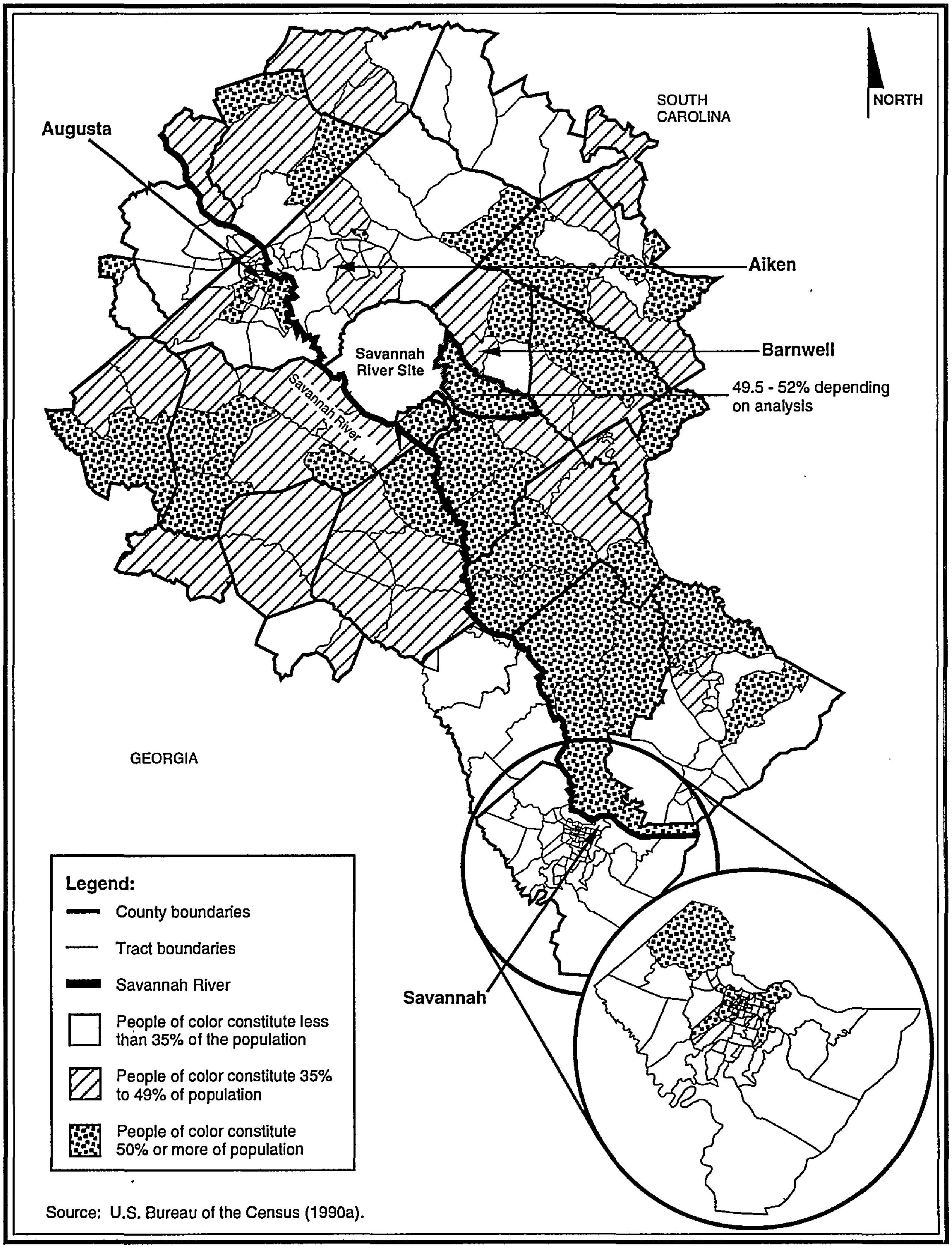
a. Source: U.S. Bureau of the Census (1990b).

b. Families with income less than the statistical poverty threshold, which in 1990 was 1989 income of $8,076 for a family of two.

Executive Order 12898 does not define minority populations. One approach is to identify communities that contain a simple majority of people of color (greater than or equal to 50 percent of the total community population). A second approach, proposed by EPA for environmental justice purposes, defines communities of people of color as those that have higher-than-average (over the region of interest) percentages of minority persons (EPA 1994). Figure 3-7 has two shading patterns to indicate census tracts where (1) people of color constitute 50 percent or more of the total population in the census tract, or (2) people of color constitute between 35 percent and 50 percent of the total population in the tract. For this analysis, DOE has adopted the second, more expansive, approach to identify people-of-color communities. TE

The combined region has 80 tracts (32.4 percent) where populations of people of color constitute 50 percent or more of the total population of the tract. In an additional 50 tracts (13.5 percent), people of color constitute between 35 and 50 percent of the population. These tracts are well distributed throughout the region, although there are more of them toward the south and in the immediate vicinities of Augusta and Savannah, Georgia.

Low-income communities are defined as those in which 25 percent or more of the population is characterized as living in poverty (EPA 1993). The U.S. Bureau of the Census defines persons in poverty as those whose income is less than a "statistical poverty threshold." This threshold is a



Source: U.S. Bureau of the Census (1990a).

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Figure 3-7. Distribution of people of color by census tracts in the SRS region of analysis.

TC

weighted average based on family size and the age of the persons in the family. The baseline threshold for the 1990 census was a 1989 income of $8,076 for a family of two.

Table 3-8 indicates that in the SRS region, more than 169,000 persons (17.0 percent of the total population) are characterized as living in poverty. In Figure 3-8, shaded census tracts identify low-income communities. In the region, 72 tracts (29.1 percent) are identified as low-income communities. These tracts are distributed throughout the region of analysis, but primarily to the south and west of the SRS. As discussed in Chapter 4, no adverse health effects are likely to occur in any offsite community, including minority and low-income communities.

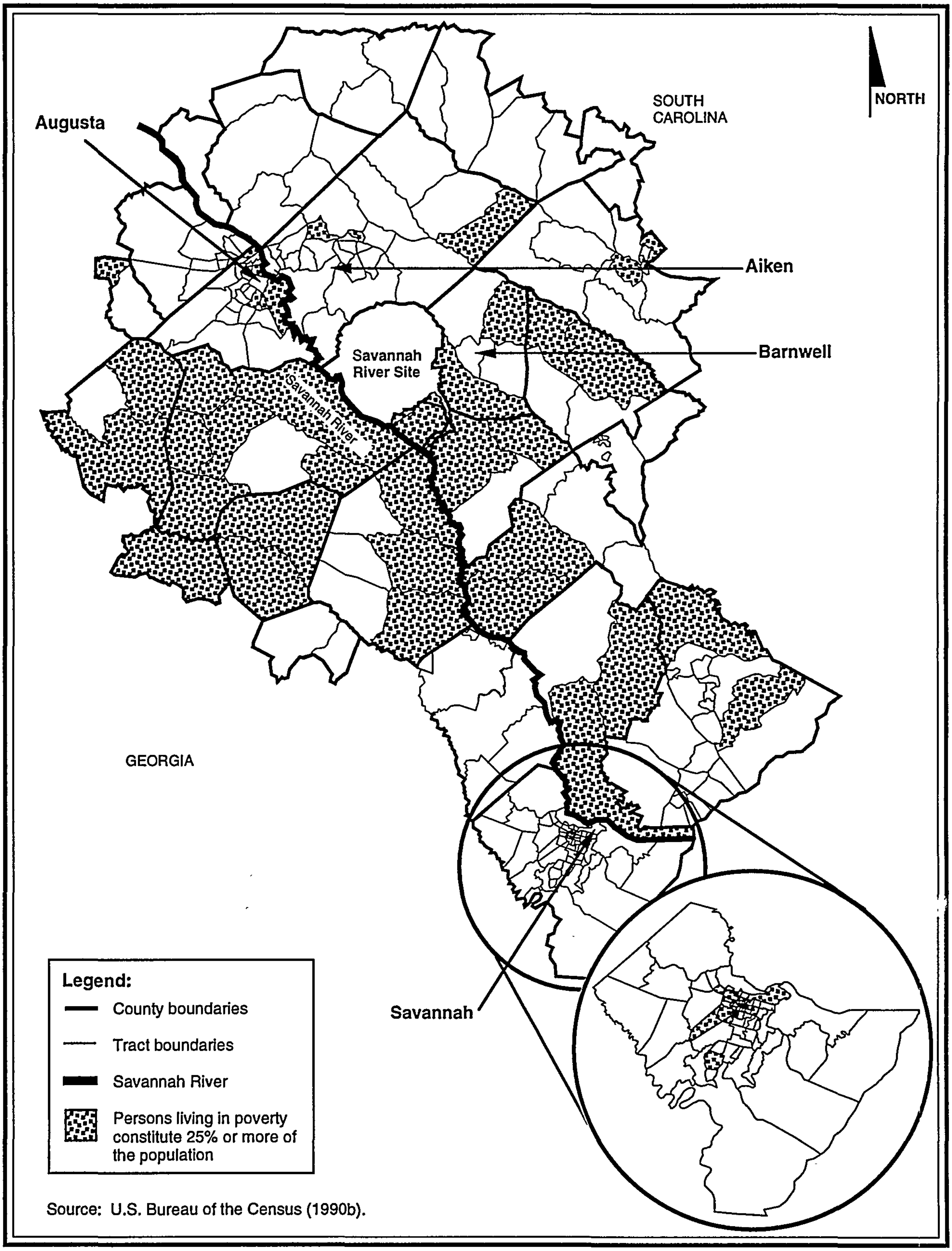
3.5 Public and Worker Health

3.5.1 PUBLIC RADIOLOGICAL HEALTH

The release of radioactivity to the environment from any nuclear facility is a sensitive issue for onsite workers and the public. Because there are many other sources of radiation in the human environment, evaluations of radioactive releases from nuclear facilities must consider all the ionizing radiation to which people are routinely exposed.

Public radiation exposure in the vicinity of the Site amounts to approximately 357 millirem per year, consisting of natural background radiation from cosmic, terrestrial, and internal body sources; radiation from medical diagnostic and therapeutic practices; radiation from weapons test fallout; radiation from consumer and industrial products; and radiation from nuclear facilities. Figure 3-9 shows the relative contributions of each source to people living in the vicinity of the Site. All radiation doses mentioned in this EIS are "effective dose equivalents"; internal exposures are reported as "committed effective dose equivalents."

Releases of radioactivity to the environment from the Site account for less than 0.1 percent of the total annual average environmental radiation dose to individuals within 80 kilometers (50 miles) of TE the Site. Natural background radiation contributes about 293 millirem per year, or 82 percent of the annual dose of 357 millirem received by an average member of the population within 80 kilometers (50 miles) of the Site. Based on national averages, medical exposure accounts for an additional 14.8 percent of the annual dose, and the combined doses from weapons test fallout, consumer and industrial products, and air travel account for about 3 percent of the total dose (NCRP 1987a).



Source: U.S. Bureau of the Census (1990b).

Figure 3-8. Low-income census tracts in the SRS region of analysis.

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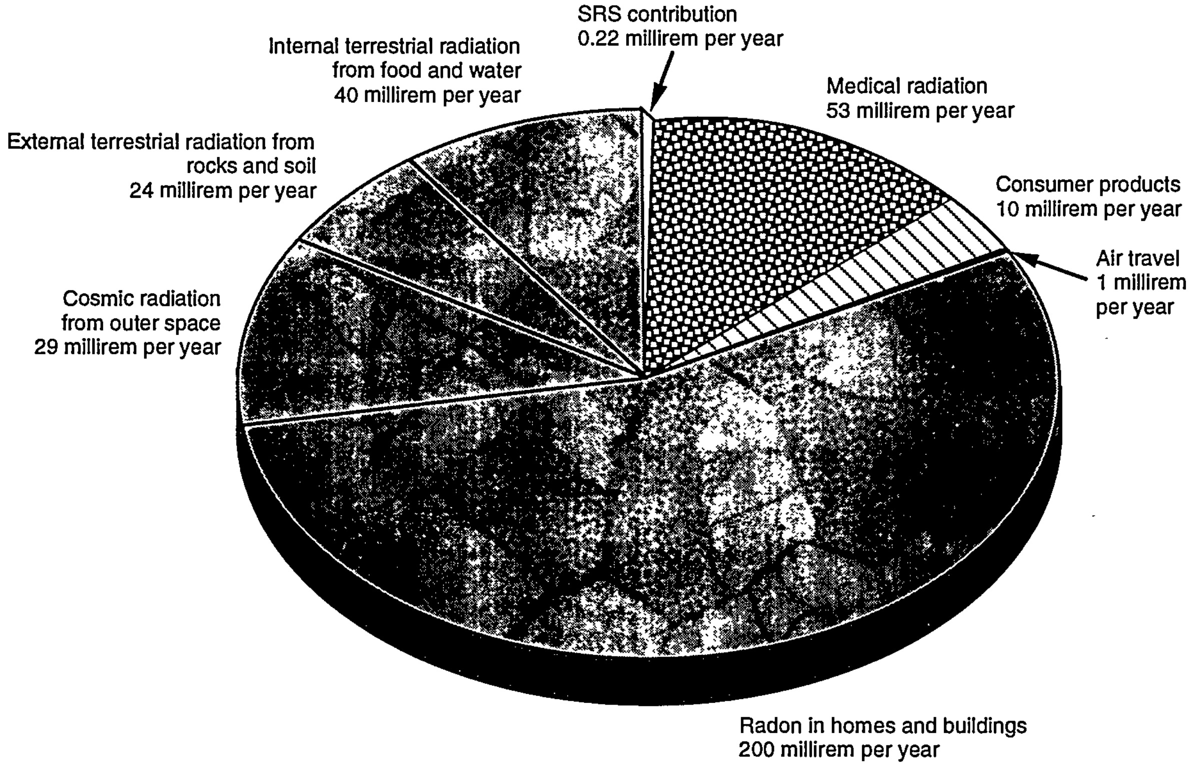

Notes:
1. Values are effective dose equivalent from NCRP (1987a) unless noted otherwise.
2. Cosmic: NCRP (1987a) reports 26 millirem per year for sea level. Multiplying value by a factor of 1.1 to correct for the altitude of 300 meters above sea level gives 29 millirem per year.
3. External terrestrial: NCRP (1987b) reports an absorbed dose rate for Augusta, Georgia, of 4 microrad per hour, which is 35 millirad per year. NCRP (1987b) uses a factor of 0.7 to convert absorbed dose in air to effective dose equivalent, so 35 x 0.7 = 24 millirem per year.
4. Value for SRS contribution is from Arnett, Karapatakis, and Mamatey (1993).

Legend:

■ Natural Background

Figure 3-9. Major sources of radiation exposure in the vicinity of the Savannah River Site.

Other nuclear facilities within 80 kilometers (50 miles) of the Site include a low-level waste burial site operated by Chem-Nuclear Systems, Inc., near the eastern SRS boundary, and the Georgia Power Company's Vogtle Electric Generating Plant, directly across the Savannah River from the Site. In addition, Carolina Metals, Inc., which is northwest of Boiling Springs in Barnwell County, processes depleted uranium. The South Carolina Department of Health and Environmental Control Annual Report for 1992 on Nuclear Facility Monitoring (SCDHEC 1992a) documents that the Chem-Nuclear and Carolina Metals facilities do not appear to influence radioactivity levels in the air, precipitation, groundwater, soil, vegetation, or external radiation, based on State measurements. Plant Vogtle began commercial operation in 1987; in 1991, releases from the plant produced a maximally exposed individual annual dose of 0.00017 rem at the plant boundary and a total population dose within an 80-kilometer (50-mile) radius of 0.057 person-rem (NRC 1994).

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In 1993, releases of radioactive material to the environment from SRS operations resulted in a maximum Site boundary individual dose from atmospheric releases of 0.11 millirem per year in the north-northwest sector around the Site, and a maximum dose from liquid releases of 0.14 millirem per year, for a maximum total annual dose at the Site boundary of 0.25 millirem. The maximum dose to downstream consumers of Savannah River water – 0.057 millirem per year – occurred to Port Wentworth public water supply users (Arnett 1994).

In 1990 the population within 80 kilometers (50 miles) of the Site was approximately 620,100. The collective effective dose equivalent to that population in 1993 was 7.6 person-rem from atmospheric releases. The 1990 population of 65,000 people using water from the Cherokee Hill Water Treatment Plant near Port Wentworth, Georgia, and the Beaufort-Jasper Water Treatment Plant near Beaufort, South Carolina, received a collective dose equivalent of 1.5 person-rem (Arnett 1994). Population statistics indicate that cancer caused 23.5 percent of the deaths in the United States in 1990 (CDC 1993). If this percentage of deaths from cancer continues, 23.5 percent of the U.S. population will contract a fatal cancer from all causes. Thus, in the population of 620,100 within 80 kilometers (50 miles) of the site, 145,700 persons would be likely to contract fatal cancers from all causes. The total population dose from the SRS of 9.1 person-rem (i.e., 7.6 person-rem from atmospheric pathways plus 1.5 person-rem from water pathways) could result in 0.0046 additional latent cancer death expected in the same population (based on 0.0005 cancer death per person-rem).

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3.5.2 PUBLIC NONRADIOLOGICAL HEALTH

The hazards associated with the alternatives described in this EIS include exposure to nonradiological chemicals in the form of air and water pollution. Table 3-5 lists ambient air quality standards and

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concentrations for selected pollutants. These standards are designed to protect the public health and welfare. The concentrations of pollutants from SRS sources, as listed in Table 3-5, are lower than the standards. Section 3.2.2 discusses water quality in the vicinity of the SRS.

3.5.3 WORKER RADIOLOGICAL HEALTH

One of the major goals of the SRS Health Protection Program is to keep worker exposures to radiation and radioactive material as low as reasonably achievable (ALARA). An effective ALARA program must balance minimizing individual worker doses with minimizing the collective dose of all workers in a given group.

TE The purpose of an ALARA program is to minimize doses from both external and internal exposure. Such a program must evaluate both doses with the goal to minimize the total effective dose equivalent. ALARA evaluations must consider individual and collective doses to ensure the minimization of both. Using many workers to perform extremely small portions of a task would reduce the individual worker doses to very low levels. However, the frequent worker changes would make the work inefficient, with the result that the total dose received by all the workers would be significantly higher than if fewer workers received slightly higher individual doses.

SRS worker doses have typically been well below DOE worker exposure limits. DOE has set administrative exposure guidelines at a fraction of the exposure limits to help enforce doses that are as low as reasonably achievable. For example, the current DOE worker exposure limit is 5 rem per year, and the 1993 SRS administrative exposure guideline was 1.5 rem per year.

Table 3-9 lists the maximum and average individual doses and the SRS collective doses from 1988 to 1993.

Workers exposed to radiation have an additional risk of 0.04 percent per person-rem of contracting a fatal cancer (NCRP 1993a). In 1993, 5,157 SRS workers received a measurable dose of radiation. TE Statistically, these workers would be likely to contract approximately 1,200 fatal cancers from all causes during their lifetimes; however, this cancer incidence rate depends on the age and sex distribution of the population. In 1993, this group received 263 person-rem and could experience as many as 0.1 additional cancer death due to their 1993 occupational radiation exposure. Continuing operation of SRS could result in as many as 0.1 additional cancer death for each year of operation, assuming future annual worker exposures continue at the 1993 level.

Table 3-9. SRS annual individual and collective radiation doses.[a]

Year	Individual dose (rem)		Site collective dose (person-rem)
	Maximum	Average[b]	
1988	2.040	0.070	864
1989	1.645	0.056	754
1990	1.470	0.056	661
1991	1.025	0.038	392
1992	1.360	0.049	316
1993	0.878	0.051	263

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a. Sources: Du Pont (1989), Petty (1993), WSRC (1991, 1992a, 1993b, 1994c). |TE

b. The average dose includes only workers who received a measurable dose during the year.

3.5.4 WORKER NONRADIOLOGICAL HEALTH AND SAFETY

Industrial hygiene and occupational health programs deal with all aspects of a worker's health and relationship with the work environment. The basic objective of an effective occupational health program is to protect employees against health hazards in their work environment. To evaluate these hazards, routine monitoring determines employee exposure levels to hazardous chemicals. Exposure limit values are the basis of most occupational health codes and standards. If an overexposure to a harmful agent does not exist, that agent generally does not create a health problem.

The Occupational Safety and Health Administration (OSHA) has established Permissible Exposure Limits (PELs) to regulate worker exposure to hazardous chemicals. These exposure limits refer to airborne concentrations of substances and represent conditions under which nearly all workers could receive repeated exposures day after day without adverse health effects.

Table 3-10 lists the estimated maximum annual concentrations of existing OSHA-regulated workplace pollutants modeled in and around the F- and H-Canyons. Virtually all nonradiological air pollutant emissions for each material evaluated in this EIS would be associated with these areas. These nonradiological concentrations are associated with the continued maintenance and storage of nuclear materials and, with the exception of nitric acid, carbon monoxide, sulfur dioxide, and nitrogen dioxide (as NO_x), should not change from current levels. Section 4.1.2 describes the incremental impacts for nitric acid, carbon monoxide, sulfur dioxide, and NO_x. Estimated concentration levels for existing OSHA-regulated workplace pollutants are less than 1 percent of the OSHA Permissible Exposure Limits, with the exception of benzene, which is 2 percent of the OSHA limit averaged over 8 hours.

Table 3-10. Estimated maximum annual concentrations (milligrams per cubic meter) of workplace pollutants regulated by the Occupational Safety and Health Administration.[a]

Pollutant	OSHA PEL[b]	Time period	Concentration
Carbon monoxide	55	8 hours	0.011
Nitrogen dioxide (as NO_x)	9	Ceiling limit[c]	0.176
Total particulates	15	8 hours	0.004
Sulfur dioxide (as SO_x)	13	8 hours	0.003
Benzene	16	Ceiling limit[c]	0.230
	3.25	8 hours	0.066
Hexane	1,800	8 hours	0.066
Nitric acid	5	8 hours	0.013
Sodium hydroxide	2	8 hours	0.0008
Toluene	1,149	Ceiling limit[c]	0.230
	766	8 hours	0.066
Xylene	440	8 hours	0.066

TE | a. Estimated maximum annual impacts to workers in and around F- and H-Canyons (WSRC 1994d).
b. Occupational Safety and Health Administration (OSHA) Permissible Exposure Limits (PEL).
c. Ceiling limits are permissible exposure limits that a facility cannot exceed at any time.

DOE has established industrial hygiene and occupational health programs for the processes covered by this EIS and across the SRS to protect the health of workers from nonradiological hazards.

CHAPTER 4. ENVIRONMENTAL IMPACTS

This analysis primarily covers the environmental impacts of proposed actions during the 10-year period from 1995 to 2004. DOE chose this period because it represents the time that it might need to make and implement decisions on the ultimate disposition of the nuclear materials under consideration in this EIS. DOE used engineering studies to identify the activities that could be required to implement each alternative, the amount of time required for each step (or "phase") of the alternative, and the annual impacts estimated to occur during each phase. (Appendix D defines each phase.) A number of assumptions were required to forecast or predict the environmental impacts that could occur during this period. To the extent practical, DOE used historic data to predict and estimate future impacts or trends. If an alternative would involve new facilities or processes, DOE extrapolated data from similar operations or facilities at the SRS. For the Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative, DOE assumed the completion of technical studies, as discussed in Chapter 2, and calculated preliminary impact estimates.

Any delays associated with implementing alternatives to process programmatic materials or to stabilize materials would result in impacts comparable to those of the No-Action Alternative involving the continued storage of the materials in their present forms and locations. Similarly, any delays during processing or stabilization operations would simply extend the period of impact at the same rate. For example, the generation of low-level radioactive waste in the form of protective clothing would result from personnel continuing their work in radiologically controlled areas.

This chapter and Appendixes D and E contain calculated or estimated impact data. The discussion of environmental factors might present data calculated to several decimal places. This does not imply that DOE predicts environmental consequences to that degree of precision. Rather, this assessment retained the number of decimal places in the calculated data to enable relative comparisons between the magnitudes of the impacts resulting from alternatives or combinations of alternatives. In some cases, the data are presented in this manner to illustrate that expected impacts would be small.

As described in Chapter 2, DOE has grouped the nuclear materials into three general categories: (1) stable, (2) programmatic, and (3) candidates for stabilization. DOE evaluated the environmental impacts of a reasonable range of alternatives for processing or stabilizing the 10 types of material (americium and curium, neptunium-237, H-Canyon uranium solutions, etc.) included in categories 2 and 3 and the impacts of continuing storage for the category 1 material. The result of this effort was the analysis of environmental impacts for 49 alternatives. Appendix D presents the annual impacts expected from each alternative, depending on the activities being performed. Appendix E presents the potential impacts from accidents.

4.1 Impacts from Normal Operations

Alternatives involving the processing or repackaging of materials would involve some temporary increase in certain environmental impacts. However, when the processing or repackaging was complete, environmental impacts probably would be smaller than the impacts experienced at present. For example, an annual reduction in radiation exposure to workers would not occur until near the end of the 10-year period. The data in Appendix D indicate that the impacts from normal operations probably would be smaller after the implementation of many of the alternatives. Appendix E indicates a similar trend for potential impacts from accidents before, during, and after the implementation of alternatives.

In its consideration of subjects to evaluate in this EIS to determine environmental impacts, DOE chose the following:

L11-3
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- Health Effects from Normal Operations, including transportation and environmental justice (Section 4.1.1)
- Air Resources (Section 4.1.2)
- Water Resources (Section 4.1.3)
- Utilities (Section 4.1.4)
- Waste Management (Section 4.1.5)
- Traffic (Section 4.1.6)

Only one alternative (Improving Storage) would require the potential construction of a new facility outside the industrialized F- and H-Areas. This facility would be for the dry storage of spent nuclear fuel (see Section 4.3 and Appendix C). The impacts associated with the construction of this new facility would result in the conversion of no more than 0.4 square kilometer (100 acres) of pine forest to industrial use. Section 4.3 contains information on the potential impacts associated with the construction of the Dry Storage Facility.

Several alternatives would require modifications to existing facilities. DOE would confine such modifications to the existing facility structure(s). For alternatives that would involve new facilities to package and store plutonium, uranium, and other materials, DOE would construct such facilities in F- or H-Area. The construction would be a warehouse or concrete vault-type structure near existing nuclear facilities. Because construction would be confined to previously disturbed and developed areas, DOE expects little or no environmental impacts in the following areas:

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- Geologic resources
- Ecological resources, including threatened or endangered species
- Cultural resources
- Aesthetics and scenic resources
- Land use

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DOE analyzed the potential impacts associated with the alternatives presented in this EIS in relation to these areas. Because the activities associated with each alternative would involve the use of existing facilities (except as noted above) within industrialized areas and the existing SRS transportation infrastructure (i.e., highways, railways), the analyses indicate that there would be little or no impact on the affected environment discussed in Chapter 3. The amount of traffic would not change from current volumes, so there should be no change in the number of vehicle-wildlife collisions. DOE does not anticipate impacts to ecological resources, surface waters, or their associated wetlands because activities would be confined to developed areas. Because estimated radiological and nonradiological emissions would be small, impacts to ecological resources would not be likely. The alternatives evaluated in this EIS would not affect endangered species because activities would not occur in areas such species inhabit.

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Because construction projects would be limited to modifications of existing facilities or construction of warehouse or vault-type facilities (i.e., not complex major nuclear facilities), DOE could use the existing SRS workforce to support these projects. Similarly, DOE would use the existing SRS workforce to implement any of the alternatives considered. The resource requirements would be effectively the same for each. As a result, DOE anticipates few, if any, socioeconomic impacts from actions proposed in this EIS.

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Tables 2-2 through 2-12 provide a direct comparison between alternatives of the environmental impacts over a 10-year period for each type of material. Tables 4-1 through 4-11 provide more detailed information on environmental data by alternative for normal operations, and Table 4-12 indicates how DOE derived this information. Section 4.2 provides similar information on accidents.

L11-19
L11-21
L11-22
L11-7

Table 4-1. Data (10-year totals) for impact analysis of various stabilization alternatives for stable material (from Table D-2).[a,b]

Factor	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)	Vitrification (F-Canyon)	Improving Storage	
Atmospheric MEI dose (rem)	1.9×10^{-5}	NA	NA	NA	NA	NA	NA	
Liquid MEI dose (rem)	8.4×10^{-6}	NA	NA	NA	NA	NA	NA	
Total MEI dose (rem)	2.7×10^{-5}	NA	NA	NA	NA	NA	NA	
Atmospheric population dose (person-rem)	1.0	NA	NA	NA	NA	NA	NA	
Liquid population dose (person-rem)	0.16	NA	NA	NA	NA	NA	NA	
Total population dose (person-rem)	1.2	NA	NA	NA	NA	NA	NA	
Offsite CO concentration (μg/m^3) 1-hour average	0	NA	NA	NA	NA	NA	NA	
Offsite CO concentration (μg/m^3) 8-hour average	0	NA	NA	NA	NA	NA	NA	
Offsite NO_x concentration (μg/m^3) annual average	5.3×10^{-3}	NA	NA	NA	NA	NA	NA	
Offsite SO_2 concentration (μg/m^3) 3-hour average	0	NA	NA	NA	NA	NA	NA	
Offsite SO_2 concentration (μg/m^3) 24-hour average	0	NA	NA	NA	NA	NA	NA	
Offsite SO_2 concentration (μg/m^3) annual average	0	NA	NA	NA	NA	NA	NA	
Offsite Gaseous Fluorides (μg/m^3) 12-hour average	9.9×10^{-3}	NA	NA	NA	NA	NA	NA	
Offsite Gaseous Fluorides (μg/m^3) 24-hour average	5.3×10^{-3}	NA	NA	NA	NA	NA	NA	
Offsite Gaseous Fluorides (μg/m^3) 1-week average	2.1×10^{-3}	NA	NA	NA	NA	NA	NA	
Offsite Gaseous Fluorides (μg/m^3) 1-month average	5.7×10^{-4}	NA	NA	NA	NA	NA	NA	L11-3
Offsite HNO_3 concentration (μg/m^3) 24-hour average	0.15	NA	NA	NA	NA	NA	NA	L11-19
Offsite HNO_3 concentration (μg/m^3) annual average	9.9×10^{-3}	NA	NA	NA	NA	NA	NA	L11-21
Onsite CO concentration (mg/m^3) 8-hour average	0	NA	NA	NA	NA	NA	NA	L11-22
Onsite NO_x concentration (mg/m^3) 1-hour average	3.4×10^{-2}	NA	NA	NA	NA	NA	NA	
Onsite SO_2 concentration (mg/m^3) 8-hour average	0	NA	NA	NA	NA	NA	NA	
Onsite HNO_3 concentration (mg/m^3) 8-hour average	2.3×10^{-3}	NA	NA	NA	NA	NA	NA	
Onsite CO_2 concentration (mg/m^3) 8-hour average	0	NA	NA	NA	NA	NA	NA	
Average number of radiation workers	410	NA	NA	NA	NA	NA	NA	
Collective worker dose (person-rem)	140	NA	NA	NA	NA	NA	NA	
Water usage (millions of liters)	16,000	NA	NA	NA	NA	NA	NA	
Electricity usage (megawatt-hours)	400,000	NA	NA	NA	NA	NA	NA	
Steam usage (millions of kilograms)	1,600	NA	NA	NA	NA	NA	NA	
Fuel usage (thousands of liters)	9,600	NA	NA	NA	NA	NA	NA	
High-level liquid waste generation (million of liters)	21	NA	NA	NA	NA	NA	NA	
Equivalent DWPF canisters	40	NA	NA	NA	NA	NA	NA	
Saltstone generation (cubic meters)	11,000	NA	NA	NA	NA	NA	NA	
TRU Waste generation (cubic meters)	20	NA	NA	NA	NA	NA	NA	
Hazardous/mixed waste generation (cubic meters)	60	NA	NA	NA	NA	NA	NA	
Low-level waste generation (cubic meters)	41,000	NA	NA	NA	NA	NA	NA	

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a. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.

b. Abbreviations: CO = carbon monoxide; CO_2 = carbon dioxide; DWPF = Defense Waste Processing Facility; $HN0_3$ = nitric acid; MEI = Maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides; SO_2 = sulfur dioxide; TRU = transuranic.

Table 4-2. Data (10-year totals) for impact analysis of various stabilization alternatives for plutonium-242 (from Tables D-3 through D-7).[a,b]

Factor	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[c]	Vitrification (F-Canyon)	Improving Storage
Atmospheric MEI dose (rem)	9.4×10^{-6}	9.2×10^{-6}	8.4×10^{-5}	NA	6.1×10^{-6}	8.4×10^{-5}	NA
Liquid MEI dose (rem)	3.7×10^{-5}	2.9×10^{-5}	4.9×10^{-6}	NA	2.4×10^{-5}	5.0×10^{-6}	NA
Total MEI dose (rem)	4.7×10^{-5}	3.8×10^{-5}	8.8×10^{-5}	NA	3.0×10^{-5}	8.9×10^{-5}	NA
Atmospheric population dose (person-rem)	0.38	0.38	3.4	NA	0.25	3.4	NA
Liquid population dose (person-rem)	0.11	0.085	0.015	NA	0.070	0.015	NA
Total population dose (person-rem)	0.49	0.47	3.4	NA	0.32	3.4	NA
Offsite CO concentration (μg/m^3) 1-hour average	1.3	2.2	4.0	NA	5.9	4.0	NA
Offsite CO concentration (μg/m^3) 8-hour average	0.31	0.44	0.95	NA	1.4	0.95	NA
Offsite NO_x concentration (μg/m^3) annual average	1.2×10^{-2}	0.14	3.3×10^{-2}	NA	0.033	0.11	NA
Offsite SO_2 concentration (μg/m^3) 3-hour average	7.5×10^{-4}	1.3×10^{-3}	2.3×10^{-3}	NA	3.4×10^{-3}	2.3×10^{-3}	NA
Offsite SO_2 concentration (μg/m^3) 24-hour average	1.7×10^{-5}	2.3×10^{-4}	5.2×10^{-4}	NA	7.7×10^{-4}	5.2×10^{-4}	NA
Offsite SO_2 concentration (μg/m^3) annual average	1.1×10^{-5}	1.6×10^{-5}	3.3×10^{-5}	NA	4.8×10^{-5}	3.3×10^{-5}	NA
Offsite Gaseous Fluorides (μg/m^3) 12-hour average	0	0.026	0	NA	7.5×10^{-6}	0.021	NA
Offsite Gaseous Fluorides (μg/m^3) 24-hour average	0	0.014	0	NA	3.0×10^{-6}	0.011	NA
Offsite Gaseous Fluorides (μg/m^3) 1-week average	0	5.5×10^{-3}	0	NA	1.6×10^{-6}	4.4×10^{-3}	NA
Offsite Gaseous Fluorides (μg/m^3) 1-month average	0	1.6×10^{-3}	0	NA	4.5×10^{-7}	1.3×10^{-3}	NA
Offsite HNO_3 concentration (μg/m^3) 24-hour average	3.2×10^{-3}	0.39	8.7×10^{-3}	NA	0.022	0.31	NA
Offsite HNO_3 concentration (μg/m^3) annual average	2.0×10^{-4}	0.026	5.5×10^{-4}	NA	1.4×10^{-3}	0.021	NA
Onsite CO concentration (mg/m^3) 8-hour average	2.1×10^{-3}	2.5×10^{-3}	6.3×10^{-3}	NA	9.3×10^{-3}	6.3×10^{-3}	NA
Onsite NO_x concentration (mg/m^3) 1-hour average	6.3×10^{-2}	0.089	1.7×10^{-2}	NA	0.018	0.072	NA
Onsite SO_2 concentration (mg/m^3) 8-hour average	3.0×10^{-6}	3.6×10^{-6}	9.2×10^{-6}	NA	1.4×10^{-5}	9.2×10^{-6}	NA
Onsite HNO_3 concentration (mg/m^3) 8-hour average	5.7×10^{-5}	6.1×10^{-3}	1.5×10^{-4}	NA	3.9×10^{-4}	4.9×10^{-3}	NA
Onsite CO_2 concentration (mg/m^3) 8-hour average	1.5×10^{-6}	1.8×10^{-6}	4.6×10^{-6}	NA	6.7×10^{-6}	4.6×10^{-6}	NA
Average number of radiation workers	134	146	55	NA	141	58	NA
Collective worker dose (person-rem)	13	87	59	NA	8.8	67	NA
Water usage (millions of liters)	1,700	3,000	2,100	NA	1,100	2,100	NA
Electricity usage (megawatt-hours)	133,000	127,000	41,000	NA	85,000	42,000	NA
Steam usage (millions of kilograms)	790	680	110	NA	510	110	NA
Fuel usage (thousands of liters)	5,000	4,300	590	NA	3,200	620	NA
High-level liquid waste generation (million of liters)	1.2	1.2	0.12	NA	0.77	0.16	NA
Equivalent DWPF canisters	20	21	2	NA	23	3	NA
Saltstone generation (cubic meters)	3,300	3,200	330	NA	2,100	420	NA
TRU Waste generation (cubic meters)	0	21	56	NA	0	61	NA
Hazardous/mixed waste generation (cubic meters)	0	0	60	NA	0	60	NA
Low-level waste generation (cubic meters)	5,600	6,500	4,300	NA	3,500	4,700	NA

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L11-19
L11-21
L-11-22

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a. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.

b. Abbreviations: CO = carbon monoxide; CO_2 = carbon dioxide; DWPF = Defense Waste Processing Facility; $HN0_3$ = nitric acid; MEI = Maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides; SO_2 = sulfur dioxide; TRU = transuranic.

c. The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.

Table 4-3. Data (10-year totals) for impact analysis of various stabilization alternatives for americium and curium (from Tables D-8 through D-14).[a,b]

Factor	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[c]	Vitrification (F-Canyon)	Improving Storage
Atmospheric MEI dose (rem)	1.5×10^{-5}	NA	5.2×10^{-5}	NA	1.8×10^{-5}	2.2×10^{-5}	NA
Liquid MEI dose (rem)	4.8×10^{-6}	NA	5.1×10^{-6}	NA	4.5×10^{-6}	2.8×10^{-6}	NA
Total MEI dose (rem)	2.0×10^{-5}	NA	5.7×10^{-5}	NA	2.2×10^{-5}	2.5×10^{-5}	NA
Atmospheric population dose (person-rem)	0.67	NA	2.3	NA	0.78	1.0	NA
Liquid population dose (person-rem)	0.030	NA	0.026	NA	0.023	0.016	NA
Total population dose (person-rem)	0.70	NA	2.3	NA	0.81	1.0	NA
Offsite CO concentration (μg/m^3) 1-hour average	0	NA	0	NA	0	0	NA
Offsite CO concentration (μg/m^3) 8-hour average	0	NA	0	NA	0	0	NA
Offsite NO_x concentration (μg/m^3) annual average	0.033	NA	0.28	NA	0.23	0.28	NA
Offsite SO_2 concentration (μg/m^3) 3-hour average	0	NA	0	NA	0	0	NA
Offsite SO_2 concentration (μg/m^3) 24-hour average	0	NA	0	NA	0	0	NA
Offsite SO_2 concentration (μg/m^3) annual average	0	NA	0	NA	0	0	NA
Offsite Gaseous Fluorides (μg/m^3) 12-hour average	6.1×10^{-3}	NA	0.052	NA	0.042	0.052	NA
Offsite Gaseous Fluorides (μg/m^3) 24-hour average	3.3×10^{-3}	NA	0.028	NA	0.023	0.028	NA
Offsite Gaseous Fluorides (μg/m^3) 1-week average	1.3×10^{-3}	NA	0.011	NA	8.9×10^{-3}	0.011	NA
Offsite Gaseous Fluorides (μg/m^3) 1-month average	3.6×10^{-4}	NA	3.1×10^{-3}	NA	2.5×10^{-3}	3.1×10^{-3}	NA
Offsite HNO_3 concentration (μg/m^3) 24-hour average	0.090	NA	0.78	NA	0.62	0.78	NA
Offsite HNO_3 concentration (μg/m^3) annual average	6.1×10^{-3}	NA	0.053	NA	0.042	0.053	NA
Onsite CO concentration (mg/m^3) 8-hour average	0	NA	0	NA	0	0	NA
Onsite NO_x concentration (mg/m^3) 1-hour average	0.021	NA	0.18	NA	0.14	0.18	NA
Onsite SO_2 concentration (mg/m^3) 8-hour average	0	NA	0	NA	0	0	NA
Onsite HNO_3 concentration (mg/m^3) 8-hour average	1.4×10^{-3}	NA	0.012	NA	9.8×10^{-3}	0.012	NA
Onsite CO_2 concentration (mg/m^3) 8-hour average	0	NA	0	NA	0	0	NA
Average number of radiation workers	209	NA	282	NA	231	152	NA
Collective worker dose (person-rem)	84	NA	320	NA	110	130	NA
Water usage (millions of liters)	7,200	NA	9,500	NA	5,700	5,200	NA
Electricity usage (megawatt-hours)	140,000	NA	181,000	NA	110,000	100,000	NA
Steam usage (millions of kilograms)	720	NA	920	NA	580	500	NA
Fuel usage (thousands of liters)	4,200	NA	5,500	NA	3,200	3,000	NA
High-level liquid waste generation (million of liters)	1.7	NA	6.9	NA	2.6	2.7	NA
Equivalent DWPF canisters	30	NA	140	NA	50	54	NA
Saltstone generation (cubic meters)	3,600	NA	18,000	NA	6,100	6,900	NA
TRU Waste generation (cubic meters)	0	NA	0	NA	0	0	NA
Hazardous/mixed waste generation (cubic meters)	0	NA	0	NA	0	0	NA
Low-level waste generation (cubic meters)	7,700	NA	9,200	NA	6,300	5,100	NA

L11-3
L11-19
L11-21
L11-22

a. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.

b. Abbreviations: CO = carbon monoxide; CO_2 = carbon dioxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = Maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides; SO_2 = sulfur dioxide; TRU = transuranic.

c. The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.

Table 4-4. Data (10-year totals) for impact analysis of various stabilization alternatives for neptunium (from Tables D-15 through D-18).[a,b]

Factor	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[c]	Vitrification (F-Canyon)	Improving Storage
Atmospheric MEI dose (rem)	1.0×10^{-5}	NA	1.4×10^{-3}	NA	2.3×10^{-4}	8.8×10^{-6}	NA
Liquid MEI dose (rem)	4.1×10^{-5}	NA	4.5×10^{-5}	NA	2.8×10^{-5}	3.2×10^{-5}	NA
Total MEI dose (rem)	5.1×10^{-5}	NA	1.4×10^{-3}	NA	2.5×10^{-4}	4.0×10^{-5}	NA
Atmospheric population dose (person-rem)	0.42	NA	56	NA	9.2	0.36	NA
Liquid population dose (person-rem)	0.12	NA	0.14	NA	0.082	0.092	NA
Total population dose (person-rem)	0.54	NA	56	NA	9.3	0.46	NA
Offsite CO concentration (μg/m^3) 1-hour average	1.8	NA	13	NA	11	13	NA
Offsite CO concentration (μg/m^3) 8-hour average	0.43	NA	3.2	NA	2.5	3.2	NA
Offsite NO_x concentration (μg/m^3) annual average	0.019	NA	0.10	NA	0.083	0.11	NA
Offsite SO_2 concentration (μg/m^3) 3-hour average	1.1×10^{-3}	NA	7.8×10^{-3}	NA	6.2×10^{-3}	7.8×10^{-3}	NA
Offsite SO_2 concentration (μg/m^3) 24-hour average	2.4×10^{-4}	NA	1.7×10^{-3}	NA	1.4×10^{-3}	1.8×10^{-3}	NA
Offsite SO_2 concentration (μg/m^3) annual average	1.5×10^{-5}	NA	1.1×10^{-4}	NA	8.8×10^{-5}	1.1×10^{-4}	NA
Offsite Gaseous Fluorides (μg/m^3) 12-hour average	0	NA	1.5×10^{-5}	NA	7.5×10^{-6}	2.1×10^{-2}	NA
Offsite Gaseous Fluorides (μg/m^3) 24-hour average	0	NA	6.0×10^{-6}	NA	3.0×10^{-6}	1.1×10^{-2}	NA
Offsite Gaseous Fluorides (μg/m^3) 1-week average	0	NA	3.2×10^{-6}	NA	1.6×10^{-6}	4.4×10^{-3}	NA
Offsite Gaseous Fluorides (μg/m^3) 1-month average	0	NA	8.9×10^{-7}	NA	4.5×10^{-7}	1.3×10^{-3}	NA
Offsite HNO_3 concentration (μg/m^3) 24-hour average	5.2×10^{-3}	NA	0.028	NA	0.022	0.311	NA
Offsite HNO_3 concentration (μg/m^3) annual average	3.3×10^{-4}	NA	1.7×10^{-3}	NA	1.4×10^{-3}	0.021	NA
Onsite CO concentration (mg/m^3) 8-hour average	2.8×10^{-3}	NA	0.021	NA	0.017	0.021	NA
Onsite NO_x concentration (mg/m^3) 1-hour average	0.010	NA	0.055	NA	0.044	0.072	NA
Onsite SO_2 concentration (mg/m^3) 8-hour average	4.2×10^{-6}	NA	3.1×10^{-5}	NA	2.5×10^{-5}	3.1×10^{-5}	NA
Onsite HNO_3 concentration (mg/m^3) 8-hour average	9.2×10^{-5}	NA	4.9×10^{-4}	NA	3.9×10^{-4}	4.9×10^{-3}	NA
Onsite CO_2 concentration (mg/m^3) 8-hour average	2.1×10^{-6}	NA	1.5×10^{-5}	NA	1.2×10^{-5}	1.5×10^{-5}	NA
Average number of radiation workers	152	NA	196	NA	162	139	NA
Collective worker dose (person-rem)	15	NA	130	NA	14	49	NA
Water usage (millions of liters)	1,900	NA	2,800	NA	1,200	2,600	NA
Electricity usage (megawatt-hours)	142,000	NA	149,000	NA	93,000	126,000	NA
Steam usage (millions of kilograms)	840	NA	760	NA	550	680	NA
Fuel usage (thousands of liters)	5,300	NA	4,700	NA	3,500	4,300	NA
High-level liquid waste generation (million of liters)	1.3	NA	4.2	NA	1.5	1.0	NA
Equivalent DWPF canisters	20	NA	37	NA	27	16	NA
Saltstone generation (cubic meters)	3,600	NA	11,000	NA	4,000	2,800	NA
TRU Waste generation (cubic meters)	0	NA	160	NA	0	0	NA
Hazardous/mixed waste generation (cubic meters)	0	NA	200	NA	0	0	NA
Low-level waste generation (cubic meters)	5,700	NA	6,400	NA	3,800	4,600	NA

L11-3
L11-19
L11-21
L11-22

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a. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.

b. Abbreviations: CO = carbon monoxide; CO_2 = carbon dioxide; DWPF = Defense Waste Processing Facility; $HN0_3$ = nitric acid; MEI = Maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides; SO_2 = sulfur dioxide; TRU = transuranic.

c. The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.

Table 4-5. Data (10-year totals) for impact analysis of various stabilization alternatives for H-Canyon plutonium-239 solutions (from Tables D-19 through D-23).[a,b]

Factor	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[c]	Vitrification (F-Canyon)	Improving Storage
Atmospheric MEI dose (rem)	9.4×10^{-6}	1.0×10^{-5}	8.7×10^{-5}	NA	2.0×10^{-3}	8.9×10^{-6}	NA
Liquid MEI dose (rem)	3.7×10^{-5}	2.8×10^{-5}	1.8×10^{-5}	NA	5.2×10^{-5}	2.9×10^{-5}	NA
Total MEI dose (rem)	4.7×10^{-5}	3.8×10^{-5}	1.1×10^{-4}	NA	2.0×10^{-3}	3.7×10^{-5}	NA
Atmospheric population dose (person-rem)	0.38	0.42	3.5	NA	81	0.37	NA
Liquid population dose (person-rem)	0.11	0.084	0.054	NA	0.16	0.084	NA
Total population dose (person-rem)	0.49	0.50	3.6	NA	81	0.45	NA
Offsite CO concentration ($\mu g/m^3$) 1-hour average	1.3	1.3	4.0	NA	11	4.0	NA
Offsite CO concentration ($\mu g/m^3$) 8-hour average	0.31	0.31	0.95	NA	2.5	0.95	NA
Offsite NO_x concentration ($\mu g/m^3$) annual average	0.012	0.14	0.033	NA	0.083	0.11	NA
Offsite SO_2 concentration ($\mu g/m^3$) 3-hour average	7.5×10^{-4}	7.5×10^{-4}	2.3×10^{-3}	NA	6.2×10^{-3}	2.3×10^{-3}	NA
Offsite SO_2 concentration ($\mu g/m^3$) 24-hour average	1.7×10^{-4}	1.7×10^{-4}	5.2×10^{-4}	NA	1.4×10^{-3}	5.2×10^{-4}	NA
Offsite SO_2 concentration ($\mu g/m^3$) annual average	1.1×10^{-5}	1.1×10^{-5}	3.3×10^{-5}	NA	8.8×10^{-5}	3.3×10^{-5}	NA
Offsite Gaseous Fluorides ($\mu g/m^3$) 12-hour average	0	0.026	0	NA	7.5×10^{-6}	0.021	NA
Offsite Gaseous Fluorides ($\mu g/m^3$) 24-hour average	0	0.014	0	NA	3.0×10^{-6}	0.011	NA
Offsite Gaseous Fluorides ($\mu g/m^3$) 1-week average	0	5.5×10^{-3}	0	NA	1.6×10^{-6}	4.4×10^{-3}	NA
Offsite Gaseous Fluorides ($\mu g/m^3$) 1-month average	0	1.6×10^{-3}	0	NA	4.5×10^{-7}	1.3×10^{-3}	NA
Offsite HNO_3 concentration ($\mu g/m^3$) 24-hour average	3.2×10^{-3}	0.39	8.7×10^{-3}	NA	0.022	0.31	NA
Offsite HNO_3 concentration ($\mu g/m^3$) annual average	2.0×10^{-4}	0.026	5.5×10^{-4}	NA	1.4×10^{-3}	0.021	NA
Onsite CO concentration (mg/m^3) 8-hour average	2.1×10^{-3}	2.1×10^{-3}	6.3×10^{-3}	NA	0.017	2.1×10^{-3}	NA
Onsite NO_x concentration (mg/m^3) 1-hour average	6.3×10^{-3}	0.089	0.017	NA	0.044	0.072	NA
Onsite SO_2 concentration (mg/m^3) 8-hour average	3.0×10^{-6}	3.0×10^{-6}	9.2×10^{-6}	NA	2.5×10^{-5}	9.1×10^{-6}	NA
Onsite HNO_3 concentration (mg/m^3) 8-hour average	5.7×10^{-5}	6.1×10^{-3}	1.5×10^{-4}	NA	3.9×10^{-4}	4.9×10^{-3}	NA
Onsite CO_2 concentration (mg/m^3) 8-hour average	1.5×10^{-6}	1.5×10^{-6}	4.6×10^{-6}	NA	1.2×10^{-5}	4.6×10^{-6}	NA
Average number of radiation workers	134	158	103	NA	206	132	NA
Collective worker dose (person-rem)	13	110	65	NA	50	53	NA
Water usage (millions of liters)	1,700	3,200	2,700	NA	2,000	2,800	NA
Electricity usage (megawatt-hours)	133,000	135,000	89,000	NA	151,000	124,000	NA
Steam usage (millions of kilograms)	790	730	390	NA	890	670	NA
Fuel usage (thousands of liters)	5,000	4,600	2,400	NA	5,700	4,200	NA
High-level liquid waste generation (million of liters)	1.2	1.3	0.55	NA	6.8	1.0	NA
Equivalent DWPF canisters	20	24	9	NA	190	17	NA
Saltstone generation (cubic meters)	3,300	3,500	1,500	NA	19,000	2,700	NA
TRU Waste generation (cubic meters)	0	32	56	NA	0	0	NA
Hazardous/mixed waste generation (cubic meters)	0	0	63	NA	0	0	NA
Low-level waste generation (cubic meters)	5,600	7,500	6,300	NA	6,400	4,800	NA

L11-3
L11-19
L11-21
L11-22

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a. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.

b. Abbreviations: CO = carbon monoxide; CO_2 = carbon dioxide; DWPF = Defense Waste Processing Facility; $HN0_3$ = nitric acid; MEI = Maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides; SO_2 = sulfur dioxide; TRU = transuranic.

c. The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.

Table 4-6. Data (10-year totals) for impact analysis of various stabilization alternatives for H-Canyon enriched uranium solutions (from Tables D-24 through D-27).[a,b]

Factor	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[c]	Vitrification (F-Canyon)	Improving Storage
Atmospheric MEI dose (rem)	1.4×10^{-5}	NA	1.7×10^{-4}	4.4×10^{-4}	1.1×10^{-5}	NA	NA
Liquid MEI dose (rem)	5.7×10^{-5}	NA	9.1×10^{-6}	1.4×10^{-5}	4.5×10^{-5}	NA	NA
Total MEI dose (rem)	7.1×10^{-5}	NA	1.8×10^{-4}	4.6×10^{-4}	5.6×10^{-5}	NA	NA
Atmospheric population dose (person-rem)	0.58	NA	6.8	18	0.46	NA	NA
Liquid population dose (person-rem)	0.16	NA	0.028	0.044	0.13	NA	NA
Total population dose (person-rem)	0.75	NA	6.8	18	0.59	NA	NA
Offsite CO concentration ($\mu g/m^3$) 1-hour average	3.9	NA	11	11	5.9	NA	NA
Offsite CO concentration ($\mu g/m^3$) 8-hour average	0.94	NA	2.5	2.5	1.4	NA	NA
Offsite NO_x concentration ($\mu g/m^3$) annual average	0.053	NA	0.083	0.083	0.053	NA	NA
Offsite SO_2 concentration ($\mu g/m^3$) 3-hour average	2.3×10^{-3}	NA	6.2×10^{-3}	6.2×10^{-3}	3.4×10^{-3}	NA	NA
Offsite SO_2 concentration ($\mu g/m^3$) 24-hour average	5.2×10^{-4}	NA	1.4×10^{-3}	1.4×10^{-3}	7.7×10^{-4}	NA	NA
Offsite SO_2 concentration ($\mu g/m^3$) annual average	3.2×10^{-5}	NA	8.8×10^{-5}	8.8×10^{-4}	3.2×10^{-5}	NA	NA
Offsite Gaseous Fluorides ($\mu g/m^3$) 12-hour average	0	NA	7.5×10^{-6}	4.2×10^{-3}	7.5×10^{-6}	NA	NA
Offsite Gaseous Fluorides ($\mu g/m^3$) 24-hour average	0	NA	3.0×10^{-6}	2.3×10^{-3}	3.0×10^{-6}	NA	NA
Offsite Gaseous Fluorides ($\mu g/m^3$) 1-week average	0	NA	1.6×10^{-6}	8.9×10^{-4}	1.6×10^{-6}	NA	NA
Offsite Gaseous Fluorides ($\mu g/m^3$) 1-month average	0	NA	4.5×10^{-7}	2.5×10^{-4}	4.5×10^{-7}	NA	NA
Offsite HNO_3 concentration ($\mu g/m^3$) 24-hour average	0.014	NA	0.022	0.062	0.022	NA	NA
Offsite HNO_3 concentration ($\mu g/m^3$) annual average	8.9×10^{-4}	NA	1.4×10^{-3}	4.2×10^{-3}	1.4×10^{-3}	NA	NA
Onsite CO concentration (mg/m^3) 8-hour average	6.3×10^{-3}	NA	0.017	0.017	9.3×10^{-3}	NA	NA
Onsite NO_x concentration (mg/m^3) 1-hour average	0.028	NA	0.044	0.044	0.028	NA	NA
Onsite SO_2 concentration (mg/m^3) 8-hour average	9.1×10^{-6}	NA	2.5×10^{-5}	2.5×10^{-5}	1.4×10^{-5}	NA	NA
Onsite HNO_3 concentration (mg/m^3) 8-hour average	2.5×10^{-4}	NA	3.9×10^{-4}	9.8×10^{-4}	3.9×10^{-4}	NA	NA
Onsite CO_2 concentration (mg/m^3) 8-hour average	4.5×10^{-6}	NA	1.2×10^{-5}	1.2×10^{-5}	6.7×10^{-6}	NA	NA
Average number of radiation workers	232	NA	55	160	246	NA	NA
Collective worker dose (person-rem)	23	NA	71	18	18	NA	NA
Water usage (millions of liters)	2,400	NA	520	640	1,800	NA	NA
Electricity usage (megawatt-hours)	180,000	NA	40,000	42,000	140,000	NA	NA
Steam usage (millions of kilograms)	1,100	NA	240	250	830	NA	NA
Fuel usage (thousands of liters)	6,800	NA	1,500	1,600	5,300	NA	NA
High-level liquid waste generation (million of liters)	1.8	NA	0.72	1.7	1.4	NA	NA
Equivalent DWPF canisters	30	NA	7	17	130	NA	NA
Saltstone generation (cubic meters)	5,000	NA	2,000	4,800	3,900	NA	NA
TRU Waste generation (cubic meters)	0	NA	0	0	0	NA	NA
Hazardous/mixed waste generation (cubic meters)	0	NA	0	0	0	NA	NA
Low-level waste generation (cubic meters)	6,300	NA	1,200	1,600	4,800	NA	NA

L11-3
L11-19
L11-21
L11-22

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a. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.

b. Abbreviations: CO = carbon monoxide; CO_2 = carbon dioxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = Maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides; SO_2 = sulfur dioxide; TRU = transuranic.

c. The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.

Table 4-7. Data (10-year totals) for impact analysis of various stabilization alternatives for plutonium and uranium stored in vaults (from Tables D-28 through D-37).[a,b]

Factor	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[c]	Vitrification (F-Canyon)	Improving Storage
Atmospheric MEI dose (rem)	4.1×10^{-6}	3.4×10^{-3}	3.4×10^{-3}	NA	3.3×10^{-3}	3.4×10^{-3}	9.9×10^{-6}
Liquid MEI dose (rem)	1.6×10^{-5}	6.6×10^{-5}	6.6×10^{-5}	NA	6.0×10^{-5}	6.6×10^{-5}	1.6×10^{-5}
Total MEI dose (rem)	2.0×10^{-5}	3.5×10^{-3}	3.5×10^{-3}	NA	3.4×10^{-3}	3.5×10^{-3}	2.6×10^{-5}
Atmospheric population dose (person-rem)	0.17	140	140	NA	140	140	0.43
Liquid population dose (person-rem)	0.046	0.22	0.22	NA	0.20	0.22	0.049
Total population dose (person-rem)	0.21	140	140	NA	140	140	0.48
Offsite CO concentration ($\mu g/m^3$) 1-hour average	1.2	18	18	NA	16	18	1.5
Offsite CO concentration ($\mu g/m^3$) 8-hour average	0.29	4.2	4.2	NA	3.9	4.2	0.35
Offsite NO_x concentration ($\mu g/m^3$) annual average	9.5×10^{-3}	0.14	0.14	NA	0.13	0.14	0.031
Offsite SO_2 concentration ($\mu g/m^3$) 3-hour average	7.2×10^{-4}	1.0×10^{-2}	1.0×10^{-2}	NA	9.5×10^{-3}	1.0×10^{-2}	8.5×10^{-4}
Offsite SO_2 concentration ($\mu g/m^3$) 24-hour average	1.6×10^{-4}	2.3×10^{-3}	2.3×10^{-3}	NA	2.1×10^{-3}	2.3×10^{-3}	1.9×10^{-4}
Offsite SO_2 concentration ($\mu g/m^3$) annual average	1.0×10^{-5}	1.5×10^{-4}	1.5×10^{-4}	NA	1.3×10^{-4}	1.5×10^{-4}	1.2×10^{-5}
Offsite Gaseous Fluorides ($\mu g/m^3$) 12-hour average	0	1.5×10^{-5}	1.5×10^{-5}	NA	1.5×10^{-5}	1.5×10^{-5}	3.8×10^{-3}
Offsite Gaseous Fluorides ($\mu g/m^3$) 24-hour average	0	6.0×10^{-6}	6.0×10^{-6}	NA	6.0×10^{-6}	6.0×10^{-6}	2.0×10^{-3}
Offsite Gaseous Fluorides ($\mu g/m^3$) 1-week average	0	3.2×10^{-6}	3.2×10^{-6}	NA	3.2×10^{-6}	3.2×10^{-6}	8.0×10^{-4}
Offsite Gaseous Fluorides ($\mu g/m^3$) 1-month average	0	8.9×10^{-7}	8.9×10^{-7}	NA	8.9×10^{-7}	8.9×10^{-7}	2.3×10^{-4}
Offsite HNO_3 concentration ($\mu g/m^3$) 24-hour average	2.5×10^{-3}	0.045	0.045	NA	0.035	0.045	0.059
Offsite HNO_3 concentration ($\mu g/m^3$) annual average	1.6×10^{-4}	2.4×10^{-3}	2.4×10^{-3}	NA	2.2×10^{-3}	2.4×10^{-3}	4.0×10^{-3}
Onsite CO concentration (mg/m^3) 8-hour average	2.0×10^{-3}	2.8×10^{-2}	0.028	NA	0.026	2.8×10^{-2}	2.3×10^{-3}
Onsite NO_x concentration (mg/m^3) 1-hour average	5.0×10^{-3}	0.076	0.076	NA	0.070	0.076	1.9×10^{-2}
Onsite SO_2 concentration (mg/m^3) 8-hour average	2.5×10^{-6}	4.1×10^{-5}	4.1×10^{-5}	NA	3.8×10^{-5}	4.1×10^{-5}	2.8×10^{-6}
Onsite HNO_3 concentration (mg/m^3) 8-hour average	4.5×10^{-5}	6.8×10^{-4}	6.8×10^{-4}	NA	6.3×10^{-4}	6.8×10^{-4}	9.3×10^{-4}
Onsite CO_2 concentration (mg/m^3) 8-hour average	1.4×10^{-6}	2.6×10^{-5}	2.6×10^{-5}	NA	1.9×10^{-5}	2.6×10^{-5}	1.7×10^{-6}
Average number of radiation workers	159	330	330	NA	336	330	158
Collective worker dose (person-rem)	140	460	460	NA	280	460	410
Water usage (millions of liters)	7,500	2,700	2,700	NA	5,900	2,700	2,000
Electricity usage (megawatt-hours)	147,000	190,000	190,000	NA	210,000	190,000	77,000
Steam usage (millions of kilograms)	130	680	700	NA	680	680	96
Fuel usage (thousands of liters)	470	4,100	4,100	NA	4,000	4,100	480
High-level liquid waste generation (million of liters)	0	8.2	8.2	NA	8.2	8.2	0
Equivalent DWPF canisters	0	61	61	NA	2,400	61	0
Saltstone generation (cubic meters)	0	22,000	22,000	NA	22,000	22,000	0
TRU Waste generation (cubic meters)	810	1,300	1,300	NA	900	1,300	1,000
Hazardous/mixed waste generation (cubic meters)	970	1,400	1,400	NA	1,100	1,400	960
Low-level waste generation (cubic meters)	19,000	24,000	24,000	NA	19,000	24,000	23,000

L11-3
L11-19
L11-21
L11-22

[a] To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.

[b] Abbreviations: CO = carbon monoxide; CO_2 = carbon dioxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = Maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides; SO_2 = sulfur dioxide; TRU = transuranic.

[c] The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.

Table 4-8. Data (10-year totals) for impact analysis of various stabilization alternatives for Mark-31 targets (from Tables D-38 through D-43).[a,b]

Factor	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[c]	Vitrification (F-Canyon)	Improving Storage	
							Traditional construction schedule	Accelerated construction schedule
Atmospheric MEI dose (rem)	1.9×10^{-6}	1.1×10^{-5}	9.7×10^{-6}	NA	1.9×10^{-5}	1.4×10^{-5}	1.9×10^{-6}	9.5×10^{-7}
Liquid MEI dose (rem)	0	3.2×10^{-7}	2.6×10^{-7}	NA	5.0×10^{-7}	4.1×10^{-7}	0	4.4×10^{-7}
Total MEI dose (rem)	1.9×10^{-6}	1.1×10^{-5}	9.9×10^{-6}	NA	1.9×10^{-5}	1.5×10^{-5}	1.9×10^{-6}	1.4×10^{-6}
Atmospheric population dose (person-rem)	0.12	0.50	0.45	NA	0.86	0.64	0.12	0.061
Liquid population dose (person-rem)	0	1.9×10^{-3}	1.6×10^{-3}	NA	2.9×10^{-3}	2.4×10^{-3}	0	0.0026
Total population dose (person-rem)	0.12	0.50	0.45	NA	0.86	0.64	0.12	0.064
Offsite CO concentration (μg/m^3) 1-hour average	0	0	0	NA	0	0	0	0
Offsite CO concentration (μg/m^3) 8-hour average	0	0	0	NA	0	0	0	0
Offsite NO_x concentration (μg/m^3) annual average	0	0.28	0.28	NA	0.23	0.34	0	0
Offsite SO_2 concentration (μg/m^3) 3-hour average	0	0	0	NA	0	0	0	0
Offsite SO_2 concentration (μg/m^3) 24-hour average	0	0	0	NA	0	0	0	0
Offsite SO_2 concentration (μg/m^3) annual average	0	0	0	NA	0	0	0	0
Offsite Gaseous Fluorides (μg/m^3) 12-hour average	0	0.052	0.052	NA	0.042	0.063	0	0
Offsite Gaseous Fluorides (μg/m^3) 24-hour average	0	0.028	0.028	NA	0.023	0.034	0	0
Offsite Gaseous Fluorides (μg/m^3) 1-week average	0	0.011	0.011	NA	8.9×10^{-3}	0.013	0	0
Offsite Gaseous Fluorides (μg/m^3) 1-month average	0	3.1×10^{-3}	3.1×10^{-3}	NA	2.5×10^{-3}	3.8×10^{-3}	0	0
Offsite HNO_3 concentration (μg/m^3) 24-hour average	0	0.78	0.78	NA	0.62	0.93	0	0
Offsite HNO_3 concentration (μg/m^3) annual average	0	0.053	0.053	NA	0.042	0.063	0	0
Onsite CO concentration (mg/m^3) 8-hour average	0	0	0	NA	0	0	0	0
Onsite NO_x concentration (mg/m^3) 1-hour average	0	0.18	0.18	NA	0.14	0.22	0	0
Onsite SO_2 concentration (mg/m^3) 8-hour average	0	0	0	NA	0	0	0	0
Onsite HNO_3 concentration (mg/m^3) 8-hour average	0	0.012	0.012	NA	9.8×10^{-3}	0.15	0	0
Onsite CO_2 concentration (mg/m^3) 8-hour average	0	0	0	NA	0	0	0	0
Average number of radiation workers	8	119	99	NA	89	137	8	27
Collective worker dose (person-rem)	14	210	180	NA	110	250	14	19
Water usage (millions of liters)	10	4,200	3,300	NA	2,400	4,600	10	60
Electricity usage (megawatt-hours)	14	64,000	51,000	NA	44,000	71,000	14	1,400
Steam usage (millions of kilograms)	10	180	150	NA	220	210	10	20
Fuel usage (thousands of liters)	14	1,000	830	NA	1,400	1,300	14	20
High-level liquid waste generation (million of liters)	1.2	2.1	1.9	NA	3.7	2.6	1.2	0.87
Equivalent DWPF canisters	28	43	41	NA	170	53	28	18
Saltstone generation (cubic meters)	3,200	5,700	5,200	NA	10,000	7,000	3,200	2,300
TRU Waste generation (cubic meters)	0	77	62	NA	0	93	0	0
Hazardous/mixed waste generation (cubic meters)	50	16	20	NA	34	16	50	35
Low-level waste generation (cubic meters)	29,000	18,000	18,000	NA	22,000	19,000	29,000	21,000

L11-3
L11-13
L11-21
L11-22

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a. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.

b. Abbreviations: CO = carbon monoxide; CO_2 = carbon dioxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = Maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides; SO_2 = sulfur dioxide; TRU = transuranic.

c. The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.

Table 4-9. Data (10-year totals) for impact analysis of various stabilization alternatives for Mark-16 and -22 fuels (from Tables D-44 through D-48).[a,b]

Factor	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[c]	Vitrification (F-Canyon)	Improving Storage	
							Traditional construction schedule	Accelerated construction schedule
Atmospheric MEI dose (rem)	5.0×10^{-7}	NA	1.7×10^{-3}	2.0×10^{-3}	3.5×10^{-5}	NA	5.0×10^{-7}	2.5×10^{-7}
Liquid MEI dose (rem)	0	NA	2.5×10^{-5}	3.1×10^{-5}	1.0×10^{-6}	NA	0	8.8×10^{-7}
Total MEI dose (rem)	5.0×10^{-7}	NA	1.7×10^{-3}	2.0×10^{-3}	3.6×10^{-5}	NA	5.0×10^{-7}	1.1×10^{-6}
Atmospheric population dose (person-rem)	0.032	NA	67	81	1.6	NA	0.032	0.016
Liquid population dose (person-rem)	0	NA	0.083	0.11	5.8×10^{-3}	NA	0	0.0051
Total population dose (person-rem)	0.032	NA	67	81	1.6	NA	0.032	0.021
Offsite CO concentration ($\mu g/m^3$) 1-hour average	0	NA	11	11	0	NA	0	0
Offsite CO concentration ($\mu g/m^3$) 8-hour average	0	NA	2.5	2.5	0	NA	0	0
Offsite NO_x concentration ($\mu g/m^3$) annual average	0	NA	0.083	0.083	0.23	NA	0	0
Offsite SO_2 concentration ($\mu g/m^3$) 3-hour average	0	NA	6.2×10^{-3}	6.2×10^{-3}	0	NA	0	0
Offsite SO_2 concentration ($\mu g/m^3$) 24-hour average	0	NA	1.4×10^{-3}	1.4×10^{-3}	0	NA	0	0
Offsite SO_2 concentration ($\mu g/m^3$) annual average	0	NA	8.8×10^{-5}	8.8×10^{-5}	0	NA	0	0
Offsite Gaseous Fluorides ($\mu g/m^3$) 12-hour average	0	NA	7.5×10^{-6}	4.2×10^{-3}	0.042	NA	0	0
Offsite Gaseous Fluorides ($\mu g/m^3$) 24-hour average	0	NA	3.0×10^{-6}	2.3×10^{-3}	0.023	NA	0	0
Offsite Gaseous Fluorides ($\mu g/m^3$) 1-week average	0	NA	1.6×10^{-6}	8.9×10^{-4}	8.9×10^{-3}	NA	0	0
Offsite Gaseous Fluorides ($\mu g/m^3$) 1-month average	0	NA	4.5×10^{-7}	2.5×10^{-4}	2.5×10^{-3}	NA	0	0
Offsite HNO_3 concentration ($\mu g/m^3$) 24-hour average	0	NA	0.022	0.062	0.62	NA	0	0
Offsite HNO_3 concentration ($\mu g/m^3$) annual average	0	NA	1.4×10^{-3}	4.2×10^{-3}	0.042	NA	0	0
Onsite CO concentration (mg/m^3) 8-hour average	0	NA	0.017	0.017	0	NA	0	0
Onsite NO_x concentration (mg/m^3) 1-hour average	0	NA	0.044	0.044	0.14	NA	0	0
Onsite SO_2 concentration (mg/m^3) 8-hour average	0	NA	2.5×10^{-5}	2.5×10^{-5}	0	NA	0	0
Onsite HNO_3 concentration (mg/m^3) 8-hour average	0	NA	3.9×10^{-4}	9.8×10^{-4}	9.8×10^{-3}	NA	0	0
Onsite CO_2 concentration (mg/m^3) 8-hour average	0	NA	1.2×10^{-5}	1.2×10^{-5}	0	NA	0	0
Average number of radiation workers	5	NA	121	114	138	NA	5	36
Collective worker dose (person-rem)	7.1	NA	200	66	220	NA	7.1	17
Water usage (millions of liters)	0	NA	1,000	1,400	4,800	NA	0	80
Electricity usage (megawatt-hours)	10	NA	79,000	83,000	89,000	NA	10	2,800
Steam usage (millions of kilograms)	10	NA	470	500	440	NA	10	30
Fuel usage (thousands of liters)	10	NA	3,000	3,100	2,800	NA	10	20
High-level liquid waste generation (million of liters)	0.57	NA	5.6	7.3	6.8	NA	0.57	0.37
Equivalent DWPF canisters	10	NA	49	68	1,000	NA	10	5
Saltstone generation (cubic meters)	1,600	NA	15,000	20,000	19,000	NA	1,600	800
TRU Waste generation (cubic meters)	0	NA	0	0	0	NA	0	0
Hazardous/mixed waste generation (cubic meters)	20	NA	22	28	44	NA	20	10
Low-level waste generation (cubic meters)	15,000	NA	16,000	20,000	32,000	NA	15,000	7,700

L11-3
L11-19
L11-21
L11-22

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a. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.

b. Abbreviations: CO = carbon monoxide; CO_2 = carbon dioxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = Maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides; SO_2 = sulfur dioxide; TRU = transuranic.

c. The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.

Table 4-10. Data (10-year totals) for impact analysis of various stabilization alternatives for other aluminum-clad targets (from Tables D-49 through D-51).[a,b]

Factor	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[c]	Vitrification (F-Canyon)	Improving Storage: Traditional construction schedule	Improving Storage: Accelerated construction schedule
Atmospheric MEI dose (rem)	1.5×10^{-7}	NA	NA	NA	1.7×10^{-4}	NA	1.5×10^{-7}	7.6×10^{-8}
Liquid MEI dose (rem)	0	NA	NA	NA	2.5×10^{-6}	NA	0	2.2×10^{-7}
Total MEI dose (rem)	1.5×10^{-7}	NA	NA	NA	1.7×10^{-4}	NA	1.5×10^{-7}	3.0×10^{-7}
Atmospheric population dose (person-rem)	9.8×10^{-3}	NA	NA	NA	6.7	NA	9.8×10^{-3}	0.0049
Liquid population dose (person-rem)	0	NA	NA	NA	8.3×10^{-3}	NA	0	0.0013
Total population dose (person-rem)	9.8×10^{-3}	NA	NA	NA	6.7	NA	9.8×10^{-3}	0.0062
Offsite CO concentration ($\mu g/m^3$) 1-hour average	0	NA	NA	NA	11	NA	0	0
Offsite CO concentration ($\mu g/m^3$) 8-hour average	0	NA	NA	NA	2.5	NA	0	0
Offsite NO_x concentration ($\mu g/m^3$) annual average	0	NA	NA	NA	0.083	NA	0	0
Offsite SO_2 concentration ($\mu g/m^3$) 3-hour average	0	NA	NA	NA	6.2×10^{-3}	NA	0	0
Offsite SO_2 concentration ($\mu g/m^3$) 24-hour average	0	NA	NA	NA	1.4×10^{-3}	NA	0	0
Offsite SO_2 concentration ($\mu g/m^3$) annual average	0	NA	NA	NA	8.8×10^{-5}	NA	0	0
Offsite Gaseous Fluorides ($\mu g/m^3$) 12-hour average	0	NA	NA	NA	7.5×10^{-6}	NA	0	0
Offsite Gaseous Fluorides ($\mu g/m^3$) 24-hour average	0	NA	NA	NA	3.0×10^{-6}	NA	0	0
Offsite Gaseous Fluorides ($\mu g/m^3$) 1-week average	0	NA	NA	NA	1.6×10^{-6}	NA	0	0
Offsite Gaseous Fluorides ($\mu g/m^3$) 1-month average	0	NA	NA	NA	4.5×10^{-7}	NA	0	0
Offsite HNO_3 concentration ($\mu g/m^3$) 24-hour average	0	NA	NA	NA	0.022	NA	0	0
Offsite HNO_3 concentration ($\mu g/m^3$) annual average	0	NA	NA	NA	1.4×10^{-3}	NA	0	0
Onsite CO concentration (mg/m^3) 8-hour average	0	NA	NA	NA	0.017	NA	0	0
Onsite NO_x concentration (mg/m^3) 1-hour average	0	NA	NA	NA	0.044	NA	0	0
Onsite SO_2 concentration (mg/m^3) 8-hour average	0	NA	NA	NA	2.5×10^{-5}	NA	0	0
Onsite HNO_3 concentration (mg/m^3) 8-hour average	0	NA	NA	NA	3.9×10^{-4}	NA	0	0
Onsite CO_2 concentration (mg/m^3) 8-hour average	0	NA	NA	NA	1.2×10^{-5}	NA	0	0
Average number of radiation workers	2	NA	NA	NA	42	NA	2	10
Collective worker dose (person-rem)	2.1	NA	NA	NA	4.5	NA	2.1	4.6
Water usage (millions of liters)	0	NA	NA	NA	80	NA	0	20
Electricity usage (megawatt-hours)	10	NA	NA	NA	5,900	NA	10	720
Steam usage (millions of kilograms)	0	NA	NA	NA	40	NA	0	10
Fuel usage (thousands of liters)	0	NA	NA	NA	220	NA	0	5
High-level liquid waste generation (million of liters)	0.14	NA	NA	NA	0.59	NA	0.14	0.09
Equivalent DWPF canisters	0	NA	NA	NA	15	NA	0	0
Saltstone generation (cubic meters)	390	NA	NA	NA	1,600	NA	390	200
TRU Waste generation (cubic meters)	0	NA	NA	NA	0	NA	0	0
Hazardous/mixed waste generation (cubic meters)	10	NA	NA	NA	4	NA	10	5
Low-level waste generation (cubic meters)	4,200	NA	NA	NA	2,300	NA	4,200	2,300

L11-3
L11-19
L11-21
L11-22

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a. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.

b. Abbreviations: CO = carbon monoxide; CO_2 = carbon dioxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = Maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides; SO_2 = sulfur dioxide; TRU = transuranic.

c. The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.

Table 4-11. Data (10-year totals) for impact analysis of various stabilization alternatives for Taiwan Research Reactor fuel and Experimental Breeder Reactor-II slugs (from Tables D-55 through D-61).[a,b,c]

Factor	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)[d]	Vitrification (F-Canyon)	Improving Storage	
							Traditional construction schedule	Accelerated construction schedule
Atmospheric MEI dose (rem)	1.4×10^{-9}	8.9×10^{-6}	8.9×10^{-6}	NA	7.1×10^{-6}	1.2×10^{-5}	1.4×10^{-9}	1.3×10^{-9}
Liquid MEI dose (rem)	3.5×10^{-6}	8.0×10^{-7}	1.7×10^{-6}	NA	2.7×10^{-6}	8.9×10^{-7}	3.5×10^{-6}	2.7×10^{-6}
Total MEI dose (rem)	3.5×10^{-6}	9.7×10^{-6}	1.1×10^{-5}	NA	9.8×10^{-6}	1.3×10^{-5}	3.5×10^{-6}	2.7×10^{-6}
Atmospheric population dose (person-rem)	5.8×10^{-5}	0.40	0.40	NA	0.32	0.54	5.8×10^{-5}	5.4×10^{-5}
Liquid population dose (person-rem)	0.010	3.3×10^{-3}	5.8×10^{-3}	NA	8.6×10^{-3}	3.8×10^{-3}	0.010	0.010
Total population dose (person-rem)	0.010	0.40	0.40	NA	0.33	0.54	0.010	0.010
Offsite CO concentration (μg/m^3) 1-hour average	0	0	0	NA	0	0	0	0
Offsite CO concentration (μg/m^3) 8-hour average	0	0	0	NA	0	0	0	0
Offsite NO_x concentration (μg/m^3) annual average	0	0.28	0.28	NA	0.23	0.34	0	0
Offsite SO_2 concentration (μg/m^3) 3-hour average	0	0	0	NA	0	0	0	0
Offsite SO_2 concentration (μg/m^3) 24-hour average	0	0	0	NA	0	0	0	0
Offsite SO_2 concentration (μg/m^3) annual average	0	0	0	NA	0	0	0	0
Offsite Gaseous Fluorides (μg/m^3) 12-hour average	0	0.052	0.052	NA	0.042	0.063	0	0
Offsite Gaseous Fluorides (μg/m^3) 24-hour average	0	0.028	0.028	NA	0.023	0.034	0	0
Offsite Gaseous Fluorides (μg/m^3) 1-week average	0	0.011	0.011	NA	8.9×10^{-3}	0.013	0	0
Offsite Gaseous Fluorides (μg/m^3) 1-month average	0	3.1×10^{-3}	3.1×10^{-3}	NA	2.5×10^{-3}	3.8×10^{-3}	0	0
Offsite HNO_3 concentration (μg/m^3) 24-hour average	0	0.78	0.78	NA	0.62	0.93	0	0
Offsite HNO_3 concentration (μg/m^3) annual average	0	0.053	0.053	NA	0.042	0.063	0	0
Onsite CO concentration (mg/m^3) 8-hour average	0	0	0	NA	0	0	0	0
Onsite NO_x concentration (mg/m^3) 1-hour average	0	0.18	0.18	NA	0.14	0.22	0	0
Onsite SO_2 concentration (mg/m^3) 8-hour average	0	0	0	NA	0	0	0	0
Onsite HNO_3 concentration (mg/m^3) 8-hour average	0	0.012	0.012	NA	9.8×10^{-3}	0.15	0	0
Onsite CO_2 concentration (mg/m^3) 8-hour average	0	0	0	NA	0	0	0	0
Average number of radiation workers	34	116	112	NA	78	132	34	52
Collective worker dose (person-rem)	14	180	180	NA	53	220	14	21
Water usage (millions of liters)	280	4,100	3,400	NA	1,200	4,400	280	220
Electricity usage (megawatt-hours)	11,000	62,000	55,000	NA	27,000	69,000	11,000	8,600
Steam usage (millions of kilograms)	90	170	180	NA	150	210	90	67
Fuel usage (thousands of liters)	60	870	850	NA	630	1,100	60	46
High-level liquid waste generation (million of liters)	0.4	1.6	1.7	NA	1.5	2.1	0.4	0.30
Equivalent DWPF canisters	0	32	32	NA	120	41	0	0
Saltstone generation (cubic meters)	110	4,200	4,300	NA	3,300	5,500	110	77
TRU Waste generation (cubic meters)	0	67	62	NA	0	82	0	0
Hazardous/mixed waste generation (cubic meters)	0	7	4	NA	0	6	0	0
Low-level waste generation (cubic meters)	1,300	12,000	9,600	NA	1,800	13,000	1,300	1,200

L11-3
L11-19
L11-21
L11-22

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a. To be conservative, DOE derived impact data without taking credit for reductions in estimated impacts based on coprocessing of materials or similar decisions that would optimize facility usage and reduce environmental impacts.

b. DOE derived the impact data for the TRR targets and the EBR-II slugs assuming it would have to stabilize all the material; this bounds the impacts in the event of additional TRR and EBR-II material failures.

c. Abbreviations: CO = carbon monoxide; CO_2 = carbon dioxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = Maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides; SO_2 = sulfur dioxide; TRU = transuranic.

d. The values in this column are preliminary estimates, subject to revision after technical studies as discussed in Chapter 2.

Table 4-12. Impact assessment methodology for normal operations.

Environmental Factor	Parameter	Method
Atmospheric MEI[a] dose (rem) Liquid MEI dose (rem) Total MEI dose (rem) Atmospheric population dose (person-rem) Liquid population dose (person-rem) Total population dose (person-rem)	Radiological air quality; radiological water quality	These numbers are derived by using the data in Appendix D for each phase of each alternative for a specific material, multiplying by the duration of that phase, and summing the individual time-weighted values for all phases of each alternative.
Offsite CO[b] concentration 1-hour average Offsite CO concentration 8-hour average Offsite NO_x[c] concentration annual average Offsite SO_2[d] concentration 3-hour average Offsite SO_2 concentration 24-hour average Offsite SO_2 concentration annual average Offsite gaseous fluorides 12-hour average Offsite gaseous fluorides 1-week average Offsite gaseous fluorides 1-month average Offsite HNO_3[e] concentration 24-hour average Offsite HNO_3 concentration annual average Onsite CO concentration 8-hour average Onsite NO_x concentration 1-hour average Onsite SO_2 concentration 8-hour average Onsite HNO_3 concentration 8-hour average Onsite CO_2 concentration 8-hour average	Nonradiological air quality	The entries for on- and offsite concentrations of air emissions represent the highest average concentrations from Appendix D for any phase of the alternative and material.
Average number of radiation workers	Worker radiological health	The average number of radiation workers is obtained by multiplying the number of workers expected to support each phase from Appendix D by the duration of that phase, and then summing the individual time-weighted values for all phases of each alternative.
Collective worker dose	Worker radiological health	Collective worker dose is calculated by multiplying the collective worker dose for each phase from Appendix D by the duration for that phase, and then summing the individual time-weighted values for all phases of each alternative.
Water usage Electricity usage Steam usage Fuel usage	Utilities	Water, electricity, steam, and fuel usage are calculated by multiplying the appropriate entry from Appendix D for the material, alternative, and phase by the duration of the phase associated with a particular utility usage, and then summing the individual time-weighted values for all phases of each alternative.

L11-7

Table 4-12. (continued).

Environmental Factor	Parameter	Method
High-level liquid waste generation Equivalent DWPF[f] canisters Saltstone generation TRU[g] waste generation Hazardous/mixed waste generation Low-level waste generation	Waste management	Waste generation entries are calculated by multiplying the appropriate entry from Appendix D for the material, alternative, and phase by the duration of the phase associated with a particular waste generation rate, and then summing the individual time-weighted values for all phases of each alternative.
Probability of additional latent cancer fatalities from total MEI dose	Public radiological health	This value is the product of the total MEI dose listed in the appropriate table in Chapter 4 by the conversion factor of 0.0005 latent cancer fatality per rem for the public.
Additional latent cancer fatalities from total population dose	Public radiological health	This value is the product of the total population dose listed in the appropriate table in Chapter 4 by the conversion factor of 0.0005 latent cancer fatality per rem for the public.
Expected cancer fatalities from all causes in worker group with average number of radiation workers	Worker radiological health	This value is the product of the average number of workers listed in the appropriate table in Chapter 4 by the conversion factor of 0.235 cancer fatality (percent of deaths caused by cancer from all causes; see Section 3.5.1).
Additional latent cancer fatalities from collective worker dose	Worker radiological health	This value is the product of the collective worker dose listed in the appropriate table in Chapter 4 by the conversion factor of 0.0004 latent cancer fatality per rem for workers.

L11-7

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a. MEI = Maximally exposed individual at the Savannah River Site.
b. CO = carbon monoxide.
c. NO_x = nitrogen oxides.
d. SO2 = sulfur dioxide.
e. HNO_3 = nitric acid.
f. DWPF = Defense Waste Processing Facility.
g. TRU = transuranic.

Table 4-12. Impact assessment methodology for normal operations.

Environmental Factor	Parameter	Method
Atmospheric MEI[a] dose (rem) Liquid MEI dose (rem) Total MEI dose (rem) Atmospheric population dose (person-rem) Liquid population dose (person-rem) Total population dose (person-rem)	Radiological air quality; radiological water quality	These numbers are derived by using the data in Appendix D for each phase of each alternative for a specific material, multiplying by the duration of that phase, and summing the individual time-weighted values for all phases of each alternative.
Offsite CO[b] concentration 1-hour average Offsite CO concentration 8-hour average Offsite NO_x[c] concentration annual average Offsite SO_2[d] concentration 3-hour average Offsite SO_2 concentration 24-hour average Offsite SO_2 concentration annual average Offsite gaseous fluorides 12-hour average Offsite gaseous fluorides 1-week average Offsite gaseous fluorides 1-month average Offsite HNO_3[e] concentration 24-hour average Offsite HNO_3 concentration annual average Onsite CO concentration 8-hour average Onsite NO_x concentration 1-hour average Onsite SO_2 concentration 8-hour average Onsite HNO_3 concentration 8-hour average Onsite CO_2 concentration 8-hour average	Nonradiological air quality	The entries for on- and offsite concentrations of air emissions represent the highest average concentrations from Appendix D for any phase of the alternative and material.
Average number of radiation workers	Worker radiological health	The average number of radiation workers is obtained by multiplying the number of workers expected to support each phase from Appendix D by the duration of that phase, and then summing the individual time-weighted values for all phases of each alternative.
Collective worker dose	Worker radiological health	Collective worker dose is calculated by multiplying the collective worker dose for each phase from Appendix D by the duration for that phase, and then summing the individual time-weighted values for all phases of each alternative.
Water usage Electricity usage Steam usage Fuel usage	Utilities	Water, electricity, steam, and fuel usage are calculated by multiplying the appropriate entry from Appendix D for the material, alternative, and phase by the duration of the phase associated with a particular utility usage, and then summing the individual time-weighted values for all phases of each alternative.

L11-7

Table 4-12. (continued).

Environmental Factor	Parameter	Method
High-level liquid waste generation Equivalent DWPF[f] canisters Saltstone generation TRU[g] waste generation Hazardous/mixed waste generation Low-level waste generation	Waste management	Waste generation entries are calculated by multiplying the appropriate entry from Appendix D for the material, alternative, and phase by the duration of the phase associated with a particular waste generation rate, and then summing the individual time-weighted values for all phases of each alternative.
Probability of additional latent cancer fatalities from total MEI dose	Public radiological health	This value is the product of the total MEI dose listed in the appropriate table in Chapter 4 by the conversion factor of 0.0005 latent cancer fatality per rem for the public.
Additional latent cancer fatalities from total population dose	Public radiological health	This value is the product of the total population dose listed in the appropriate table in Chapter 4 by the conversion factor of 0.0005 latent cancer fatality per rem for the public.
Expected cancer fatalities from all causes in worker group with average number of radiation workers	Worker radiological health	This value is the product of the average number of workers listed in the appropriate table in Chapter 4 by the conversion factor of 0.235 cancer fatality (percent of deaths caused by cancer from all causes; see Section 3.5.1).
Additional latent cancer fatalities from collective worker dose	Worker radiological health	This value is the product of the collective worker dose listed in the appropriate table in Chapter 4 by the conversion factor of 0.0004 latent cancer fatality per rem for workers.

L11-7

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a. MEI = Maximally exposed individual at the Savannah River Site.
b. CO = carbon monoxide.
c. NO_X = nitrogen oxides.
d. SO2 = sulfur dioxide.
e. HNO_3 = nitric acid.
f. DWPF = Defense Waste Processing Facility.
g. TRU = transuranic.

4.1.1 HEALTH EFFECTS FROM NORMAL OPERATIONS

This section discusses estimated radiological and nonradiological health effects to SRS workers and the public from all the alternatives for the stabilization of nuclear materials during normal operations, which are planned activities (e.g., sampling, maintenance); it does not include the impacts of potential accidents, which are discussed in Section 4.2. This discussion represents health effects as additional lifetime latent cancer fatalities likely to occur in the general population around the SRS and in the population of workers that would be associated with the alternatives. TE

4.1.1.1 Radiological Health Effects from Facility Operations

DOE expects minimal worker and public health impacts from the radiological consequences of managing SRS nuclear materials under any of the alternatives, including No Action. However, some of the alternatives could result in increased impacts that would not occur until after the 10-year period due to facility availability or technology development. The maximum additional lifetime latent cancer fatality for the public for any alternative and any material would be 0.07 for processing to metal, processing to oxide, processing and storage for vitrification (DWPF) or vitrification in F-Canyon of plutonium and uranium stored in vaults. The maximum additional lifetime latent cancer fatality for radiation workers (0.18) would apply for the plutonium and uranium stored in vaults under the Processing to Metal, Processing to Oxide, and Vitrification (F-Canyon) Alternatives. L11-18

DOE based its calculations of health effects on (1) the dose to the hypothetical maximally exposed individual (MEI) in the public; (2) the collective dose to the population around the SRS (approximately 620,000 people); (3) the collective dose to all workers in the affected groups; and (4) the dose to the maximally exposed worker. The collective population doses include the dose from airborne releases (Section 4.1.2) and the dose from the use of the Savannah River for drinking water, recreation, and as a source of food (Section 4.1.3). The estimated worker doses are based on past operating experience and the projected durations for implementing the alternative actions (WSRC 1995a). For the maximally exposed worker, DOE assumes that no worker would receive an annual dose greater than 0.8 rem for any alternative because the SRS uses 0.8 rem as an administrative limit for normal operations (i.e., personnel receiving an annual dose at that level are normally assigned other duties in nonradiation areas). Therefore, DOE assumes that the maximally exposed worker could receive as much as 8 rem over the 10-year period regardless of the alternative. It is unlikely that the same worker would receive the maximum annual dose every year for 10 years; however, the 8-rem total dose is an upper limit that produces the maximum probability that the individual worker could contract a fatal cancer of 3 in 1,000. Tables 4-1 through 4-11 list these doses (with the exception of the maximum worker dose) for each material and each alternative applicable to each material.

The principal potential human health effect from exposure to low levels of radiation is cancer. For the purpose of this analysis, radiological carcinogenic effects are expressed as the number of fatal cancers for populations and the maximum probability of death of a maximally exposed individual. In addition to latent cancer fatalities, other health effects could result from environmental and occupational exposures to radiation. These effects include nonfatal cancers among the exposed population and genetic effects in subsequent generations. To enable comparisons with fatal cancer risk, the International Commission on Radiological Protection (ICRP 1991) suggested the use of detriment weighting factors that consider the curability rate of nonfatal cancers and the reduced quality of life associated with each type of nonfatal cancer and each type of heredity effect. The International Commission on Radiological Protection recommended risk factors for the general public of 0.0001 per person-rem for nonfatal cancers and 0.00013 per person-rem for hereditary effects. Both of these values are approximately a factor of 4 lower than the risk factors for fatal cancer. Therefore, this EIS presents estimated effects of radiation only in terms of latent cancer fatalities, because that is the health effect of most concern from exposure to radiation.

L11-18 From the collective (total) radiological doses, DOE calculated estimates of additional lifetime latent cancer fatalities using the conversion factors of 0.0004 additional latent cancer fatality per rem for workers and 0.0005 additional latent cancer fatality per rem for the public recommended by the National Council on Radiation Protection and Measurements (NCRP 1993b). The value for the public is greater than that for workers because the public consists of all age groups including children, while the worker population consists only of adults. The effects on the maximally exposed individual in the public and the maximally exposed worker were calculated not as a number of additional latent cancer fatalities but as the additional lifetime probability of contracting a fatal cancer. The same conversion factors of 0.0004 latent cancer fatality per rem for workers and 0.0005 latent cancer fatality per rem for the public were used for these probability calculations. Tables 4-13 through 4-23 list the results of these radiological health effects calculations for each material. Each of these tables compares the radiological health effects associated with each alternative applicable to a specific material. In addition to the dose values in Tables 4-1 through 4-11 and the risk values in Tables 4-13 through 4-23, Appendix D provides annual radiological data for each phase of each alternative for each material.

Table 4-13. Comparison of potential 10-year radiological health effects of the alternatives for stable material (from Table 4-1).

Factor	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)	Vitrification (F-Canyon)	Improving Storage
Probability of additional latent cancer fatalities from total MEI[a] dose	1 in 100 million	NA[b]	NA	NA	NA	NA	NA
Additional latent cancer fatalities from total population dose[c]	0.00060	NA	NA	NA	NA	NA	NA
Expected cancer fatalities from all causes in worker group with average number of radiation workers[d]	96	NA	NA	NA	NA	NA	NA
Additional latent cancer fatalities from collective worker dose[e]	0.056	NA	NA	NA	NA	NA	NA

L11-3

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a. MEI = Maximally exposed individual based on dose at the SRS boundary, including doses from atmospheric and liquid releases.

b. NA = Not applicable.

c. Based on dose to all people within 80 kilometers (50 miles) from atmospheric releases and to people using the Savannah River for drinking water and recreation, and as a source of food.

d. Based on average number of radiation workers in the involved work groups for the years in which worker exposure occurred.

e. Based on collective dose to all workers involved with the specific operation.

Table 4-14. Comparison of potential 10-year radiological health effects of the alternatives for plutonium-242 (from Table 4-2).

Factor	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)	Vitrification (F-Canyon)	Improving Storage
Probability of additional latent cancer fatalities from total MEI[a] dose	2 in 100 million	2 in 100 million	4 in 100 million	NA[b]	2 in 100 million	4 in 100 million	NA
Additional latent cancer fatalities from total population dose[c]	0.00025	0.00024	0.0017	NA	0.00016	0.0017	NA
Expected cancer fatalities from all causes in worker group with average number of radiation workers[d]	31	34	13	NA	33	14	NA
Additional latent cancer fatalities from collective worker dose[e]	0.0052	0.035	0.024	NA	0.0035	0.027	NA

L11-3

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a. MEI = Maximally exposed individual based on dose at the SRS boundary, including doses from atmospheric and liquid releases.

b. NA = Not applicable.

c. Based on dose to all people within 80 kilometers (50 miles) from atmospheric releases and to people using the Savannah River for drinking water and recreation, and as a source of food.

d. Based on average number of radiation workers in the involved work groups for the years in which worker exposure occurred.

e. Based on collective dose to all workers involved with the specific operation.

Table 4-15. Comparison of potential 10-year radiological health effects of the alternatives for americium and curium (from Table 4-3).

Factor	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)	Vitrification (F-Canyon)	Improving Storage
Probability of additional latent cancer fatalities from total MEI[a] dose	1 in 100 million	NA[b]	3 in 100 million	NA	1 in 100 million	1 in 100 million	NA
Additional latent cancer fatalities from total population dose[c]	0.00035	NA	0.0012	NA	0.00041	0.00050	NA
Expected cancer fatalities from all causes in worker group with average number of radiation workers[d]	49	NA	66	NA	54	36	NA
Additional latent cancer fatalities from collective worker dose[e]	0.034	NA	0.13	NA	0.044	0.052	NA

L11-3

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a. MEI = Maximally exposed individual based on dose at the SRS boundary, including doses from atmospheric and liquid releases.

b. NA = Not applicable.

c. Based on dose to all people within 80 kilometers (50 miles) from atmospheric releases and to people using the Savannah River for drinking water and recreation, and as a source of food.

d. Based on average number of radiation workers in the involved work groups for the years in which worker exposure occurred.

e. Based on collective dose to all workers involved with the specific operation.

Table 4-16. Comparison of potential 10-year radiological health effects of the alternatives for neptunium (from Table 4-4).

Factor	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)	Vitrification (F-Canyon)	Improving Storage
Probability of additional latent cancer fatalities from total MEI[a] dose	3 in 100 million	NA[b]	7 in 10 million	NA	1 in 10 million	2 in 100 million	NA
Additional latent cancer fatalities from total population dose[c]	0.00027	NA	0.028	NA	0.0047	0.00023	NA
Expected cancer fatalities from all causes in worker group with average number of radiation workers[d]	36	NA	46	NA	38	33	NA
Additional latent cancer fatalities from collective worker dose[e]	0.0060	NA	0.052	NA	0.0056	0.020	NA

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a. MEI = Maximally exposed individual based on dose at the SRS boundary, including doses from atmospheric and liquid releases.

b. NA = Not applicable.

c. Based on dose to all people within 80 kilometers (50 miles) from atmospheric releases and to people using the Savannah River for drinking water and recreation, and as a source of food.

d. Based on average number of radiation workers in the involved work groups for the years in which worker exposure occurred.

e. Based on collective dose to all workers involved with the specific operation.

L11-3

Table 4-17. Comparison of potential 10-year radiological health effects of the alternatives for H-Canyon plutonium-239 solutions (from Table 4-5).

Factor	Continuing Storage (No Action	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)	Vitrification (F-Canyon)	Improving Storage
Probability of additional latent cancer fatalities from total MEI[a] dose	2 in 100 million	2 in 100 million	6 in 100 million	NA[b]	1 in 1 million	2 in 100 million	NA
Additional latent cancer fatalities from total population dose[c]	0.00025	0.00025	0.0018	NA	0.041	0.00023	NA
Expected cancer fatalities from all causes in worker group with average number of radiation workers[d]	31	37	24	NA	48	31	NA
Additional latent cancer fatalities from collective worker dose[e]	0.0052	0.044	0.026	NA	0.020	0.021	NA

L11-3

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a. MEI = Maximally exposed individual based on dose at the SRS boundary, including doses from atmospheric and liquid releases.
b. NA = Not applicable.
c. Based on dose to all people within 80 kilometers (50 miles) from atmospheric releases and to people using the Savannah River for drinking water and recreation, and as a source of food.
d. Based on average number of radiation workers in the involved work groups for the years in which worker exposure occurred.
e. Based on collective dose to all workers involved with the specific operation.

Table 4-18. Comparison of potential 10-year radiological health effects of the alternatives for H-Canyon enriched uranium solution (from Table 4-6).

Factor	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)	Vitrification (F-Canyon)	Improving Storage
Probability of additional latent cancer fatalities from total MEI[a] dose	4 in 100 million	NA[b]	9 in 100 million	2 in 10 million	3 in 100 million	NA	NA
Additional latent cancer fatalities from total population dose[c]	0.00038	NA	0.0034	0.009	0.00030	NA	NA
Expected cancer fatalities from all causes in worker group with average number of radiation workers[d]	55	NA	13	38	58	NA	NA
Additional latent cancer fatalities from collective worker dose[e]	0.0092	NA	0.028	0.0072	0.0072	NA	NA

L11-3

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a. MEI = Maximally exposed individual based on dose at the SRS boundary, including doses from atmospheric and liquid releases.

b. NA = Not applicable.

c. Based on dose to all people within 80 kilometers (50 miles) from atmospheric releases and to people using the Savannah River for drinking water and recreation, and as a source of food.

d. Based on average number of radiation workers in the involved work groups for the years in which worker exposure occurred.

e. Based on collective dose to all workers involved with the specific operation.

Table 4-19. Comparison of potential 10-year radiological health effects of the alternatives for plutonium and uranium stored in vaults (from Table 4-7).

Factor	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)	Vitrification (F-Canyon)	Improving Storage
Probability of additional latent cancer fatalities from total MEI[a] dose	1 in 100 million	2 in 1 million	2 in 1 million	NA[b]	2 in 1 million	2 in 1 million	1 in 100 million
Additional latent cancer fatalities from total population dose[c]	0.00011	0.070	0.070	NA	0.070	0.070	0.00024
Expected cancer fatalities from all causes in worker group with average number of radiation workers[d]	37	78	78	NA	79	78	37
Additional latent cancer fatalities from collective worker dose[e]	0.056	0.18	0.18	NA	0.11	0.18	0.16

L11-3

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a. MEI = Maximally exposed individual based on dose at the SRS boundary, including doses from atmospheric and liquid releases.

b. NA = Not applicable.

c. Based on dose to all people within 80 kilometers (50 miles) from atmospheric releases and to people using the Savannah River for drinking water and recreation, and as a source of food.

d. Based on average number of radiation workers in the involved work groups for the years in which worker exposure occurred.

e. Based on collective dose to all workers involved with the specific operation.

Table 4-20. Comparison of potential 10-year radiological health effects of the alternatives for Mark-31 targets (from Table 4-8).

							Improving Storage	
Factor	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)	Vitrification (F-Canyon)	Traditional construction schedule	Accelerated construction schedule
Probability of additional latent cancer fatalities from total MEI[a] dose	1 in 1 billion	6 in 1 billion	5 in 1 billion	NA[b]	1 in 100 million	8 in 1 billion	1 in 1 billion	7 in 10 billion
Additional latent cancer fatalities from total population dose[c]	0.000060	0.00025	0.00023	NA	0.00043	0.00032	0.000060	0.000032
Expected cancer fatalities from all causes in worker group with average number of radiation workers[d]	2	28	23	NA	21	32	2	6
Additional latent cancer fatalities from collective worker dose[e]	0.0056	0.084	0.072	NA	0.044	0.10	0.0056	0.0076

L11-3

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a. MEI = Maximally exposed individual based on dose at the SRS boundary, including doses from atmospheric and liquid releases.

b. NA = Not applicable.

c. Based on dose to all people within 80 kilometers (50 miles) from atmospheric releases and to people using the Savannah River for drinking water and recreation, and as a source of food.

d. Based on average number of radiation workers in the involved work groups for the years in which worker exposure occurred.

e. Based on collective dose to all workers involved with the specific operation.

Table 4-21. Comparison of potential 10-year radiological health effects of the alternatives for Mark-16 and -22 fuels (from Table 4-9).

Factor	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)	Vitrification (F-Canyon)	Improving Storage: Traditional construction schedule	Improving Storage: Accelerated construction schedule
Probability of additional latent cancer fatalities from total MEI[a] dose	3 in 10 billion	NA[b]	9 in 10 million	1 in 1 million	2 in 100 million	NA	3 in 10 billion	6 in 10 billion
Additional latent cancer fatalities from total population dose[c]	0.000016	NA	0.034	0.041	0.00080	NA	0.000016	0.000011
Expected cancer fatalities from all causes in worker group with average number of radiation workers[d]	1	NA	28	27	32	NA	1	8
Additional latent cancer fatalities from collective worker dose[e]	0.0028	NA	0.080	0.026	0.088	NA	0.0028	0.0068

L11-3

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a. MEI = Maximally exposed individual based on dose at the SRS boundary, including doses from atmospheric and liquid releases.

b. NA = Not applicable.

c. Based on dose to all people within 80 kilometers (50 miles) from atmospheric releases and to people using the Savannah River for drinking water and recreation, and as a source of food.

d. Based on average number of radiation workers in the involved work groups for the years in which worker exposure occurred.

e. Based on collective dose to all workers involved with the specific operation.

Table 4-22. Comparison of potential 10-year radiological health effects of the alternatives for other aluminum-clad targets (from Table 4-10).

Factor	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)	Vitrification (F-Canyon)	Improving Storage	
							Traditional construction schedule	Accelerated construction schedule
Probability of additional latent cancer fatalities from total MEI[a] dose	8 in 100 billion	NA[b]	NA	NA	9 in 100 million	NA	8 in 100 billion	2 in 10 billion
Additional latent cancer fatalities from total population dose[c]	0.0000049	NA	NA	NA	0.0034	NA	0.0000049	0.0000031
Expected cancer fatalities from all causes in worker group with average number of radiation workers[d]	0.5	NA	NA	NA	10	NA	0.5	2.0
Additional latent cancer fatalities from collective worker dose[e]	0.00084	NA	NA	NA	0.0018	NA	0.00084	0.0018

L11-3

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a. MEI = Maximally exposed individual based on dose at the SRS boundary, including doses from atmospheric and liquid releases.

b. NA = Not applicable.

c. Based on dose to all people within 80 kilometers (50 miles) from atmospheric releases and to people using the Savannah River for drinking water and recreation, and as a source of food.

d. Based on average number of radiation workers in the involved work groups for the years in which worker exposure occurred.

e. Based on collective dose to all workers involved with the specific operation.

Table 4-23. Comparison of potential 10-year radiological health effects of the alternatives for Taiwan Research Reactor fuel and Experimental Breeder Reactor-II slugs (from Table 4-11).

							Improving Storage	
Factor	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Processing and Storage for Vitrification (DWPF)	Vitrification (F-Canyon)	Traditional construction schedule	Accelerated construction schedule
Probability of additional latent cancer fatalities from total MEI[a] dose	2 in 1 billion	5 in 1 billion	6 in 1 billion	NA	5 in 1 billion	7 in 1 billion	2 in 1 billion	1 in 1 billion
Additional latent cancer fatalities from total population dose[c]	0.000005	0.0002	0.0002	NA	0.00017	0.00027	0.000005	0.000005
Expected cancer fatalities from all causes in worker group with average number of radiation workers[d]	8	27	26	NA	18	31	8	12
Additional latent cancer fatalities from collective worker dose[e]	0.0056	0.072	0.072	NA	0.021	0.088	0.0056	0.0084

L11-3

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a. MEI = Maximally exposed individual based on dose at the SRS boundary, including doses from atmospheric and liquid releases.

b. NA = Not applicable.

c. Based on dose to all people within 80 kilometers (50 miles) from atmospheric releases and to people using the Savannah River for drinking water and recreation, and as a source of food.

d. Based on average number of radiation workers in the involved work groups for the years in which worker exposure occurred.

e. Based on collective dose to all workers involved with the specific operation.

The estimated number of latent cancer fatalities in the public listed in Tables 4-13 through 4-23 can be compared to the number of latent cancer fatalities (145,700) expected in the public around the SRS from all causes (as discussed in Section 3.5.1). Similarly, the estimated number of latent cancer fatalities in the worker population can be compared to the number of latent cancer fatalities in the worker population from all causes. Because this value would vary from alternative to alternative (due to the differing numbers of workers involved), Tables 4-13 through 4-23 list estimates of the number of cancer deaths expected in the worker population from all causes (based on the average number of radiation workers listed in Tables 4-1 through 4-11 and the estimated normal cancer fatality rate of 23.5 percent; see Section 3.5.1). L11-18

Typically, estimated radiological health effects on the public would be smallest for the Continuing Storage (No-Action) Alternative, followed by those for Improving Storage (if applicable). The radiological health effects for other alternatives would vary from one material to another, and no specific alternative would produce consistently good or bad comparative effects. The relative radiological health effects for the total dose to the population for the various alternatives for each material would be equal to the combination of those discussed in Sections 4.1.2.1 and 4.1.3.1.

The Continuing Storage Alternative would not consistently produce the lowest worker radiological health effects, as it did for the general population, because large numbers of workers would be required to maintain the various materials and the related systems necessary for safe continued storage. Analysis of the data listed in Tables 4-13 through 4-23 shows no consistent trends of worker radiological health effects with any of the alternatives, although a combination of the Processing to Metal and Processing to Oxide Alternatives would result in the highest potential worker radiological health effects. Estimates of health effects for the alternatives are influenced by several factors, as discussed in Section 2.4. An evaluation of the relative radiological health effects associated with the collective worker dose for the various alternatives for each material resulted in the following observations: L11-3

- Plutonium-242 - Worker health effects associated with Processing to Metal, Processing to Oxide, and Vitrification (F-Canyon) would be similar to each other but higher than those from the Continuing Storage Alternative because there would be doses associated with post-stabilization storage in each of these alternatives. The Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would appear to have the lowest effect because of the relatively short time required to transfer material to the waste storage tanks, and because the impacts of vitrification would not be included in the 10-year total.

- Americium and curium - Estimated worker radiological health effects from the Processing to Oxide Alternative would be higher than those from Vitrification (F-Canyon) because the operation time for the separation facilities would be longer for Processing to Oxide. Health effects from the Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would be less because of the relatively short time required to transfer this material to the waste tanks. Continuing Storage would have the smallest health effect for this material because the extra doses received from maintaining the present storage of material would not be great enough to offset doses from conversion for the other alternatives.

- Neptunium - Processing to Oxide would produce the highest worker radiological health effects because of the relatively long time that would be required to convert the material to an oxide. The radiological health effects from Vitrification (F-Canyon) would be higher than those from the Continuing Storage Alternative primarily because of the dose associated with the post-stabilization storage of the product. The Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would appear to have the lowest effect because of the relatively short time required to transfer the material to the waste storage tanks, and because vitrification would occur after the 10-year period.

- H-Canyon plutonium-239 solutions - Worker health effects from the Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would be larger than the effects from Continuing Storage because of the relatively long time that would be required to transfer the material to the waste tank. Doses from actual vitrification in DWPF would occur after the 10-year period and would further increase the dose from this alternative. The radiological health effects associated with Processing to Metal, Processing to Oxide, and Vitrification (F-Canyon) would be larger than those associated with Continuing Storage because of the doses contributed by operation of the separations facilities and the storage of the stabilized material. Of these alternatives, health effects from the Processing to Metal Alternative would be the highest because of the operation of F-Canyon and FB-Line to convert the solution to metal. L11-3

- H-Canyon enriched uranium solutions - Worker health effects associated with the Processing to Oxide Alternative would be highest because of the relatively long time required to convert the material to oxide in the Uranium Solidification Facility. Health effects from Continuing Storage would be higher than those from the Blending Down to Low Enriched Uranium Alternative because of the higher dose associated with maintaining storage in comparison to the dose from converting the material and storing the low-enriched uranium. Radiological health effects to workers from the Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would be equal to the effects from Blending Down to Low Enriched Uranium because the dose associated with transferring the material to the waste tank would be relatively low and because vitrification and its impacts would not occur during the 10-year study period.

- Plutonium and uranium stored in vaults - Worker radiological health effects for the Processing to Metal, Vitrification (F-Canyon), Improving Storage, and Processing to Oxide Alternatives would be approximately the same. Health effects associated with Processing to Metal, Vitrification (F-Canyon), and Processing to Oxide would be slightly higher than those from the other two because of the higher dose involved with conversion. Health effects associated with Processing and Storage for Vitrification in the Defense Waste Processing Facility would appear to be less than the other alternatives because vitrification and its impacts in the DWPF would not occur during the 10-year period.

- L11-3 Mark-31 targets - Worker radiological health effects would be highest for the Vitrification (F-Canyon) Alternative due to the dose received during the activities associated with vitrification. Health effects for Processing and Storage for Vitrification in the Defense Waste Processing Facility would appear to be lower than those for Vitrification (F-Canyon), Processing to Metal, and Processing to Oxide because the vitrification would not occur during the 10-year period. The health effects from Processing and Storage for Vitrification in the Defense Waste Processing Facility would be greater than those associated with Improving Storage and Continuing Storage because of the doses associated with transferring the material to the waste tanks. Radiological health effects associated with the Improving Storage Alternative would be similar to those for Continuing Storage because a new Dry Storage Facility might not be available until near the end of the 10-year period, resulting in Improving Storage being equivalent to Continuing Storage. If construction were completed sooner, the Improving Storage (Accelerated Construction Schedule) Alternative could result in slightly higher impacts.

- Mark-16 and -22 fuels - Worker radiological health effects associated with the Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would be the highest because a longer period would be required to convert this material to a form compatible with eventual vitrification than that required for Mark-31 fuel, for example. Health effects associated with Processing to Oxide also would be relatively high due to the dose contributed by activities that would be required to convert the uranium to oxide. Radiological health effects associated with the Improving Storage Alternative would be similar to those for Continuing Storage because a Dry Storage Facility might not be available until near the end of the 10-year period, as discussed for the Mark-31 targets. If construction were completed sooner, the Improving Storage (Accelerated Construction Schedule) Alternative could result in slightly higher impacts.

- Other aluminum-clad targets - Worker radiological health effects would be highest for the Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative because this is the only alternative for this material that would involve operation of the separations facilities. Radiological health effects associated with the Improving Storage Alternative would be similar to those for Continuing Storage because a Dry Storage Facility might not be available until near the end of the 10-year period, as discussed for the Mark-31 targets. If construction were completed sooner, the Improving Storage (Accelerated Construction Schedule) Alternative could result in slightly higher impacts. L11-3

- Taiwan Research Reactor and Experimental Breeder Reactor-II fuel - The Processing to Metal, Processing to Oxide, and Vitrification (F-Canyon) Alternatives would have the greatest worker radiological impacts and would be similar in magnitude because of the use of the same facilities for comparable durations of time. The Processing and Storage for Vitrification in the Defense Waste Processing Alternative would have smaller impacts because DOE would store the material for several years before transferring it to the waste tanks. The No-Action and Improving Storage Alternatives would have similar worker impacts because DOE would store the material for most or all of the 10-year period.

4.1.1.2 Nonradiological Health Effects from Facility Operations TE

DOE evaluated the range of chemicals to which the public and workers would be exposed due to SRS nuclear material management activities, and expects minimal public and worker health impacts from nonradiological health effects. Sections 4.1.2.2 and 4.1.3.2 discuss the offsite chemical concentrations from air emissions and liquid discharges, respectively. TE DOE estimated the worker impacts using the EPA Industrial Source Complex Short Term No. 2 Model to calculate concentrations in and around work areas (Hunter 1995a,b), compared them to the Occupational Safety and Health Administration (OSHA) Permissible Exposure Limits (PELs) or ceiling limits for protecting worker health, and concluded that all impacts are well below the limits. TC

OSHA limits (29 CFR Part 1910.1000) are time-weighted average concentrations that a facility cannot exceed during a prescribed duration of a 40-hour week. The facility cannot exceed OSHA ceiling concentrations during any part of the workday. These exposure limits refer to airborne concentrations of substances and represent conditions under which nearly all workers could be exposed day after day without adverse health effects. However, because of the wide variation in individual susceptibility, a small percentage of workers could experience discomfort from some substances at concentrations at or below the permissible limit. Table 4-24 summarizes the values of Permissible Exposure Limits that DOE compared to the data in Tables 4-1 through 4-11. Appendix D provides the detailed material- and alternative-specific analysis.

Table 4-24. Permissible Exposure Limits (milligrams per cubic meter) of nonradiological air pollutants regulated by the Occupational Safety and Health Administration.[a]

Pollutant	Averaging Time	OSHA PEL[b]
Carbon monoxide	8 hours	55
Nitrogen oxides	1 hours	9[c]
Sulfur dioxide	8 hours	13
Carbon dioxide	8 hours	9,000
Nitric acid	8 hours	5

a. Source: 29 CFR Part 1910.1000.
b. Occupational Safety and Health Administration (OSHA) Permissible Exposure Limit (PEL).
c. OSHA ceiling limit not to be exceeded at any time during the workday; modeled 1-hour concentrations are listed for comparison to ceiling limits.

TC

4.1.1.3 Impacts from Routine Transportation of Radioactive Material

DOE calculated transportation impacts arising from two types of shipments: the shipments of the nuclear materials directly associated with an alternative (plutonium-242, neptunium, Mark-31 targets, etc.) and the shipments of radioactive waste associated with an alternative (transuranic waste, mixed waste, saltstone, and low-level waste).

Tables 4-25 and 4-26 list the results of the analysis performed to estimate the normal or incident-free radiological transportation impacts associated with each radioactive material, radioactive waste category, and alternative. The impacts are quantified as excess latent cancer fatalities that could be observed in the exposed individual or population during the 10-year period. The analysis was limited to onsite movements because none of the alternatives included offsite transportation. If DOE decided in the future that offsite transportation was necessary, it would prepare separate National Environmental Policy Act documentation.

Table 4-25. Latent cancer fatalities resulting from routine shipment of nuclear materials.

Material	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blend Down to Low Enriched Uranium	Vitrification (F-Canyon)	Processing and Storage for Vitrification (DWPF)	Improving storage	
							Traditional construction schedule	Accelerated construction schedule
Stable material	NT[a]	NA[b]	NA	NA	NA	NA	NA	NA
Plutonium-242	NT			NA		NT	NA	NA
Uninvolved worker		7.20×10^{-19}	5.64×10^{-11}		5.64×10^{-11}			
Onsite population		4.76×10^{-15}	6.32×10^{-8}		6.32×10^{-8}			
Involved workers		2.57×10^{-15}	7.75×10^{-6}		7.75×10^{-6}			
Americium and curium		NA		NA		NT	NA	NA
Uninvolved worker	5.18×10^{-9}(c)		7.78×10^{-9}		7.78×10^{-9}			
Onsite population	4.43×10^{-7}(c)		4.04×10^{-6}		4.04×10^{-6}			
Involved workers	4.34×10^{-6}(c)		2.69×10^{-5}		2.69×10^{-5}			
Neptunium	NT	NA		NA		NT	NA	NA
Uninvolved worker			6.63×10^{-9}		6.99×10^{-10}			
Onsite population			5.65×10^{-6}		7.91×10^{-7}			
Involved workers			5.54×10^{-4}		2.58×10^{-6}			
H-Canyon plutonium-239 solutions	NT			NA		NT	NA	NA
Uninvolved worker		5.28×10^{-12}	2.18×10^{-10}		5.28×10^{-12}			
Onsite population		8.40×10^{-8}	2.47×10^{-7}		8.40×10^{-8}			
Involved workers		7.84×10^{-9}	2.99×10^{-5}		7.84×10^{-9}			
H-Canyon enriched uranium solutions	NT	NA	NT		NA	NT	NA	NA
Uninvolved worker				2.88×10^{-9}				
Onsite population				3.82×10^{-6}				
Involved workers				2.00×10^{-5}				
Plutonium and uranium stored in vaults				NA				NA
Uninvolved worker	1.24×10^{-11}(d)	1.38×10^{-9}	1.38×10^{-9}		1.38×10^{-9}	1.38×10^{-9}	1.24×10^{-11}	
Onsite population	1.39×10^{-8}(d)	1.56×10^{-6}	1.56×10^{-6}		1.56×10^{-6}	1.56×10^{-6}	1.39×10^{-8}	
Involved workers	2.54×10^{-6}(d)	1.87×10^{-4}	1.87×10^{-4}		1.87×10^{-4}	1.87×10^{-4}	2.54×10^{-6}	
Mark-31 targets	NT			NA				
Uninvolved worker		5.96×10^{-8}	5.96×10^{-8}		5.96×10^{-8}	5.96×10^{-8}	5.96×10^{-8}	5.96×10^{-8}
Onsite population		5.10×10^{-6}	5.10×10^{-6}		5.10×10^{-6}	5.10×10^{-6}	5.10×10^{-6}	5.10×10^{-6}
Involved workers		5.00×10^{-5}	5.00×10^{-5}		5.00×10^{-5}	5.00×10^{-5}	5.00×10^{-5}	5.00×10^{-5}

L11-3

Table 4-25. (continued).

Material	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blend Down to Low Enriched Uranium	Vitrification (F-Canyon)	Processing and Storage for Vitrification (DWPF)	Improving storage: Traditional construction schedule	Improving storage: Accelerated construction schedule
Mark-16 and -22 fuels	NT	NA			NA			
Uninvolved worker			2.28×10^{-7}	2.28×10^{-7}		2.28×10^{-7}	2.28×10^{-7}	2.13×10^{-7}
Onsite population			3.92×10^{-5}	3.92×10^{-5}		3.92×10^{-5}	3.92×10^{-5}	1.82×10^{-5}
Involved workers			2.88×10^{-4}	2.88×10^{-4}		2.88×10^{-4}	2.88×10^{-4}	1.78×10^{-4}
Other aluminum-clad targets	NT	NA	NA	NA	NA			
Uninvolved worker						1.04×10^{-8}	1.04×10^{-8}	1.04×10^{-8}
Onsite population						8.08×10^{-6}	8.08×10^{-6}	8.08×10^{-6}
Involved workers						8.69×10^{-6}	8.69×10^{-6}	8.69×10^{-6}
TRR/EBR-II[e] fuel	NT			NA				
Uninvolved worker		1.08×10^{-10}	1.08×10^{-10}		1.08×10^{-10}	1.08×10^{-10}	1.08×10^{-10}	1.08×10^{-10}
Onsite population		3.59×10^{-6}	3.59×10^{-6}		3.59×10^{-6}	3.59×10^{-6}	3.59×10^{-6}	3.59×10^{-6}
Involved workers		2.25×10^{-5}	2.25×10^{-5}		2.25×10^{-5}	2.25×10^{-5}	2.25×10^{-5}	2.25×10^{-5}

L11-3

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a. NT - No transportation associated with the material for the given alternative.
b. NA - Alternative not applicable for the given material.
c. Under this alternative, the americium and curium targets and slugs in P-Reactor Basin are planned to be moved to the Receiving Basin for Offsite Fuel.
d. Under this alternative, the plutonium-238 solids in various site locations are planned to be moved to a vault in HB-Line.
e. TRR - Taiwan Research Reactor; EBR - Experimental Breeder Reactor.

Table 4-26. Latent cancer fatalities resulting from routine shipment of radioactive waste.[a]

Material	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blend Down to Low Enriched Uranium	Vitrification (F-Canyon)	Processing and Storage for Vitrification (DWPF)	Improving storage: Traditional construction schedule	Improving storage: Accelerated construction schedule
Stable material		NA	NA	NA	NA	NA	NA	NA
Uninvolved worker	1.58×10^{-7}							
Onsite population	3.06×10^{-4}							
Involved workers	1.10×10^{-2}							
Plutonium-242				NA			NA	NA
Uninvolved worker	2.69×10^{-8}	2.95×10^{-8}	1.48×10^{-8}		1.63×10^{-8}	1.70×10^{-8}		
Onsite population	5.23×10^{-5}	5.72×10^{-5}	2.87×10^{-5}		3.17×10^{-5}	3.29×10^{-5}		
Involved workers	1.69×10^{-3}	1.90×10^{-3}	1.11×10^{-3}		1.21×10^{-3}	1.06×10^{-3}		
Americium and curium		NA		NA			NA	NA
Uninvolved worker	3.42×10^{-8}		8.20×10^{-8}		3.64×10^{-8}	3.33×10^{-8}		
Onsite population	6.63×10^{-5}		1.59×10^{-4}		7.06×10^{-5}	6.44×10^{-5}		
Involved workers	2.23×10^{-3}		4.10×10^{-3}		1.96×10^{-4}	1.83×10^{-3}		
Neptunium		NA		NA			NA	NA
Uninvolved worker	2.82×10^{-8}		5.50×10^{-8}		2.24×10^{-8}	2.36×10^{-8}		
Onsite population	5.46×10^{-5}		1.07×10^{-5}		4.34×10^{-5}	4.57×10^{-5}		
Involved workers	1.74×10^{-3}		2.84×10^{-3}		1.40×10^{-3}	1.33×10^{-3}		
H-Canyon plutonium-239 solutions				NA				NA
Uninvolved worker	2.69×10^{-8}	3.35×10^{-8}	2.43×10^{-8}		2.27×10^{-8}	7.67×10^{-8}		
Onsite population	5.22×10^{-5}	6.50×10^{-5}	4.71×10^{-5}		4.40×10^{-5}	1.49×10^{-4}		
Involved workers	1.69×10^{-3}	2.17×10^{-3}	1.70×10^{-3}		1.43×10^{-3}	3.55×10^{-3}		
H-Canyon enriched uranium solutions		NA			NA		NA	NA
Uninvolved worker	3.42×10^{-8}		9.67×10^{-9}	1.93×10^{-8}		2.63×10^{-8}		
Onsite population	6.63×10^{-5}		1.87×10^{-5}	3.74×10^{-5}		5.10×10^{-5}		
Involved workers	2.04×10^{-3}		4.99×10^{-4}	8.93×10^{-4}		1.56×10^{-3}		
Plutonium and uranium stored in vaults				NA				NA
Uninvolved worker	6.95×10^{-8}	1.57×10^{-7}	1.57×10^{-7}		1.57×10^{-7}	1.37×10^{-7}	8.28×10^{-8}	
Onsite population	1.35×10^{-4}	3.04×10^{-4}	3.04×10^{-4}		3.04×10^{-4}	2.66×10^{-4}	1.60×10^{-4}	
Involved workers	5.23×10^{-3}	9.11×10^{-3}	9.11×10^{-3}		9.11×10^{-3}	7.67×10^{-3}	6.22×10^{-3}	

L11-3

Table 4-26. (continued).

Material	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blend Down to Low Enriched Uranium	Vitrification (F-Canyon)	Processing and Storage for Vitrification (DWPF)	Improving storage	
							Traditional construction schedule	Accelerated construction schedule
Mark-31 targets				NA				
Uninvolved worker	9.80×10^{-8}	7.24×10^{-8}	7.08×10^{-8}		7.95×10^{-8}	9.72×10^{-8}	9.80×10^{-8}	7.09×10^{-8}
Onsite population	1.90×10^{-4}	1.40×10^{-4}	1.37×10^{-4}		1.54×10^{-4}	1.88×10^{-4}	1.90×10^{-4}	1.38×10^{-4}
Involved workers	7.28×10^{-3}	4.92×10^{-3}	4.87×10^{-3}		5.31×10^{-3}	6.33×10^{-3}	7.28×10^{-3}	5.27×10^{-3}
Mark-16 and -22 fuels		NA			NA			
Uninvolved worker	5.05×10^{-8}		9.39×10^{-8}	1.21×10^{-7}		1.55×10^{-7}	5.05×10^{-8}	2.59×10^{-8}
Onsite population	9.79×10^{-5}		1.82×10^{-4}	2.35×10^{-4}		3.00×10^{-4}	9.79×10^{-5}	5.02×10^{-5}
Involved workers	3.76×10^{-3}		5.43×10^{-3}	6.29×10^{-3}		9.68×10^{-3}	3.76×10^{-3}	1.93×10^{-3}
Other aluminum-clad targets	1.40×10^{-8}	NA	NA	NA	NA	1.18×10^{-8}	1.40×10^{-8}	7.62×10^{-9}
Uninvolved worker	2.71×10^{-5}					2.29×10^{-5}	2.71×10^{-5}	1.48×10^{-5}
Onsite population	1.05×10^{-3}					7.21×10^{-4}	1.05×10^{-3}	5.72×10^{-4}
Involved workers								
TRR/EBR-II[b] fuel				NA				
Uninvolved worker	4.35×10^{-9}	4.95×10^{-8}	4.25×10^{-8}		5.66×10^{-8}	1.54×10^{-8}	4.28×10^{-9}	3.88×10^{-9}
Onsite population	8.44×10^{-6}	9.60×10^{-5}	8.24×10^{-5}		1.10×10^{-4}	2.99×10^{-5}	8.30×10^{-6}	7.52×10^{-6}
Involved workers	3.27×10^{-4}	3.33×10^{-3}	2.76×10^{-3}		3.71×10^{-3}	7.81×10^{-4}	3.22×10^{-4}	2.94×10^{-4}

L11-3

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a. LCF are summed by receptor type and by material.

b. TRR - Taiwan Research Reactor; EBR - Experimental Breeder Reactor.

DOE used MICROSHIELD© (Negin and Worku 1993) and RADTRAN (Neuhauser and Kanipe 1992) to estimate potential health effects due to the transportation of radioactive materials. MICROSHIELD is a computer program designed to calculate external dose rates from radioactive sources based on radioisotopic information and shielding configuration. DOE used results from MICROSHIELD to calculate external doses for involved workers based on estimates of the number of shipments, number of involved workers, and worker responsibilities (proximity to shipping containers). To determine the dose to the onsite population and to an uninvolved worker, DOE entered the results from MICROSHIELD in RADTRAN, which was configured with site-specific demographic information. DOE then applied the risk factor of 0.0004 latent cancer fatality per rem to the dose estimates to generate the information in Tables 4-25 and 4-26.

The receptors for transportation analysis are defined as follows:

TC

- Uninvolved Worker - The SRS employee who is not assigned to the transportation activity, but, as a casual observer along the normal transportation route, could receive radiation exposure from the normal shipment.

- Onsite Population - The collective SRS employee population not assigned to the transportation activity that could receive external or internal radiation exposure from normal and accident shipments. Approximately 7,000 SRS employees could be exposed to routine shipments and as many as 6,000 could be exposed to radiation in the event of an accident.

- Involved Workers - The collective SRS employee population assigned to the transportation activity (i.e., transport crew and package handlers) that could receive external radiation exposure from normal shipments. This analysis assumed that a group of eight workers (per shipment) would be involved in the transportation and handling of the waste packages.

Table 4-25 lists estimated excess latent cancer fatalities associated with incident-free transport by material and alternative for the three receptor groups. The number of excess latent cancer fatalities varies by alternative for five of the 11 materials. The following paragraphs discuss these differences by material:

L11-3

- Plutonium-242 - Under the Processing to Metal Alternative, plutonium-242 would be transported in a liquid form (prestabilization) that would produce lower external dose rates than the oxide form (post-stabilization) transported under the Processing to Oxide and

Vitrification (F-Canyon) Alternatives. Excess latent cancer fatalities would vary by alternative as the external dose rates varied.

- Americium and curium - Under the Continuing Storage (No-Action) Alternative, only the targets stored in the P-Reactor disassembly basin would be transported to support consolidation efforts, resulting in a lower number of excess latent cancer fatalities than the Processing to Oxide and Vitrification (F-Canyon) Alternatives, which would require the transportation of a greater number of targets.

- Neptunium - Neptunium oxide (post-stabilization), which would be transported under the Processing to Oxide Alternative, would produce an external dose rate greater than the rate that would be produced by the neptunium solution (prestabilization) transported under the Vitrification (F-Canyon) Alternative. Excess latent cancer fatalities would vary by alternative as the external dose rates varied.

- H-Canyon plutonium solutions - Plutonium oxide, which would be transported under the Processing to Oxide Alternative (after stabilization), would produce an external dose rate greater than the rate that would be produced by the plutonium solution (prestabilization) transported under the Vitrification (F-Canyon) and Processing to Metal Alternatives. The resulting excess latent cancer fatalities would be greater for the Processing to Oxide Alternative due to the higher external dose rate. L11-3

- Plutonium and uranium stored in vaults - Under the Continuing Storage and Improving Storage Alternatives, only solids enriched in plutonium-238 would be transported to support consolidation efforts. Because only one material would be transported, the number of excess latent cancer fatalities would be lower than those for alternatives under which all materials would be transported.

For the materials discussed in this EIS that are not listed above, the impacts under the different alternatives would be the same because of similar shipping parameters (e.g., type of shipment, location, number of workers).

Table 4-25 lists impacts of material shipment according to material and alternative and expressed as latent cancer fatalities. The maximum impact to the uninvolved worker and onsite population would be attributable to the transport of Mark-16 and -22 fuels, with 0.000000228 and 0.0000392 latent cancer fatalities occurring among uninvolved workers and the onsite population, respectively. For involved workers the maximum health impact would be 0.000554 latent cancer fatality due to the transport of neptunium for the Processing to Oxide Alternative.

Table 4-26 lists estimated excess latent cancer fatalities associated with the incident-free (i.e., no accidents) transport of radioactive waste by material category and alternative for three receptor groups. Such fatalities would increase as the volume of waste increased. The number of excess latent cancer fatalities resulting from the varying volumes of radioactive waste would vary by alternative. Table 4-26 lists the human health impacts associated with the transport of all the waste forms generated from each alternative.

L11-3

The incident-free impacts would be greater for waste handling than for the materials listed in Table 4-25 due to the large volume of waste to be shipped. Neither Table 4-25 nor Table 4-26 lists impacts for the offsite population or the maximally exposed individual, because incident-free conditions would not release radioactive material to the public. However, the public could be exposed to radiation as a result of a transportation accident, as discussed in Section 4.2.2.

The data listed in Tables 4-25 and 4-26 support the expectation that the potential health effects caused by 10-year normal transport activities under any alternative would be a small fraction of the potential effects attributable to other routine SRS activities.

4.1.1.4 Environmental Justice Assessment

This EIS examines whether communities of people of color or low income (as defined in Section 3.4.3) could be recipients of disproportionately high and adverse human health and environmental impacts. Even though DOE expects little or no adverse health impacts from any of the alternatives, it analyzed whether there would be "disproportionately high and adverse human health or environmental effects [of these alternatives] on minority populations or low-income populations" (Executive Order 12898). Figures 3-7 and 3-8 show communities of people of color and low income by census tract. This section discusses predicted average radiation doses received by individuals in those communities and compares them to the predicted per capita doses that could be received in the other communities in the 80-kilometer (50-mile) region. This section also discusses impacts of doses that could be received in the downstream communities from liquid effluents from all alternatives, and also discusses potential impacts from nonradiological pollutants.

TC

Figure 4-1 shows a wheel with 22.5-degree sectors and concentric rings from 16 to 80 kilometers (10 to 50 miles) at 16-kilometer (10-mile) intervals. A fraction of the total population dose was

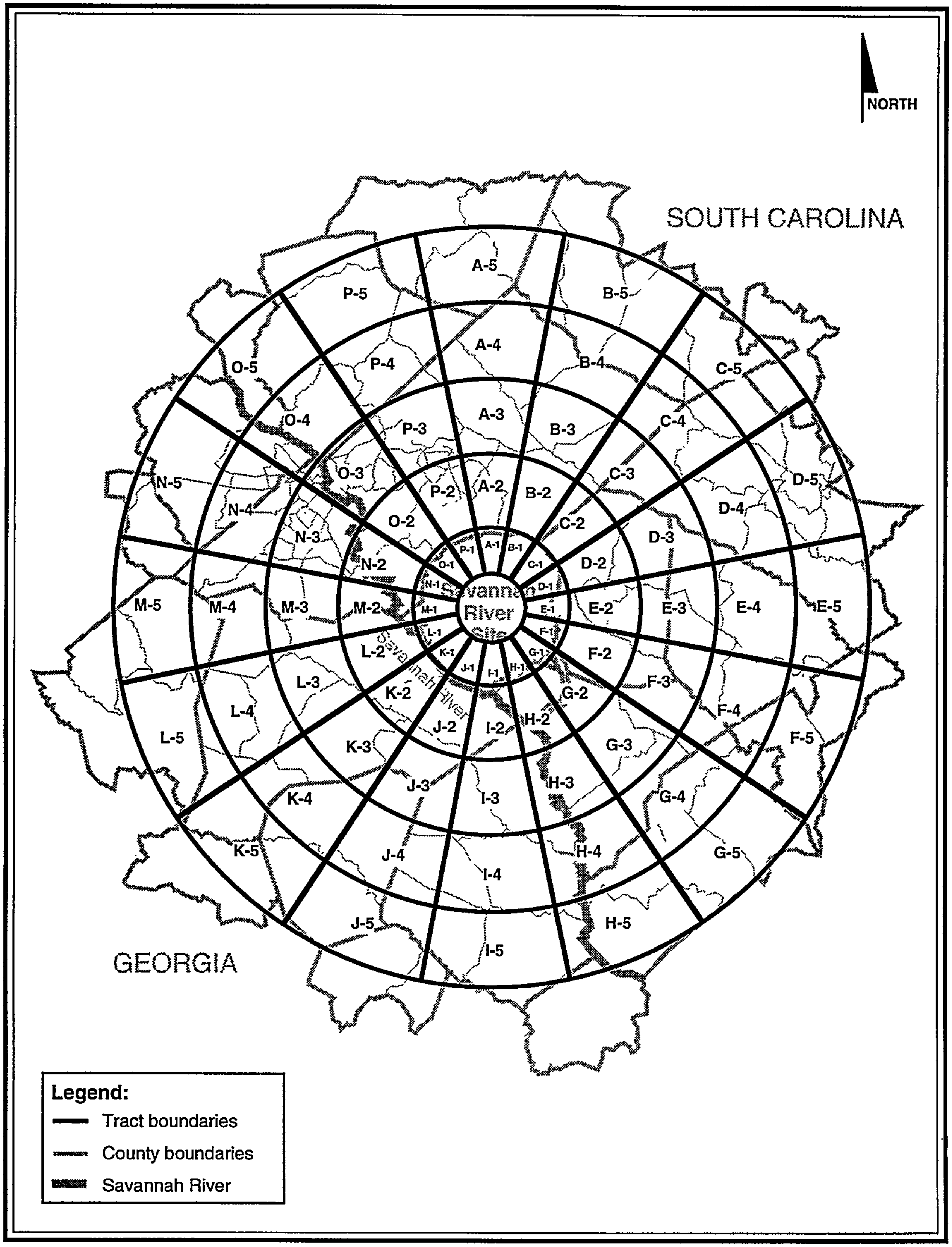

Figure 4-1. Annular sectors around the Savannah River Site.

calculated for each sector (Table 4-27), the sector wheel was laid over the census tract map, and each tract was assigned to a sector. For this analysis, if a tract fell in more than one sector, it was assigned to the sector with the largest value. TE

DOE analyzed the impacts by comparing the per capita dose received by each type of community to the other types of communities within a defined region. To eliminate the possibility that impacts to a low-population community close to the SRS with a high dose per person would be diluted and masked by including it with a high-population community farther from the SRS, the analysis made comparisons within a series of concentric circles, the radii of which increase in 16-kilometer (10-mile) increments.

To determine the radiation dose received per person in each type of community, the number of people in each tract was multiplied by that tract's dose value to obtain a total population dose for each tract. These population doses for each type of community were summed over each concentric circle and divided by the total community population to obtain a community per capita dose for each circular area. Because the per capita dose for communities (Table 4-27) would be constant for all alternatives, the relative differences in impacts between any identified communities also would remain constant for all alternatives. Thus, Figure 4-2 and Table 4-28 show the distribution of per capita dose to types of communities within the 80-kilometer region. As shown in Figure 4-2, this analysis indicates that atmospheric releases would not disproportionately affect communities of people of color (population equal to or greater than 35 percent of the total population) or low income (equal to or greater than 25 percent of the total population) in the 80-kilometer (50-mile) region; that is, when the per capita doses are compared horizontally in Figure 4-2, the per capita doses do not vary greatly. TC

For illustrative purposes, DOE used a 10-year total population dose of 1 person-rem to prepare Figure 4-2 and its supporting data in Table 4-28. For any other population dose, the per capita dose for identified communities can be determined by multiplying that population dose by the numbers in Table 4-28.

Section 4.1.3 discusses predicted potential doses to the offsite maximally exposed individual and to the downstream population from exposure to water resources. Those doses reflect people using the Savannah River for drinking water, sports, and food (fish). Because the identified communities in the areas downstream from the SRS are well distributed, there would be no disproportionate impacts among people of color or low-income communities. TE

TE

Table 4-27. Annular sector factors for local dose evaluations.

Sector[a]	Fraction of total population dose in sector[b] 1 (8-16 km)	2 (16-32 km)	3 (32-48 km)	4 (48-64 km)	5 (64-80 km)	Fraction of total population dose received by average person in sector[b] 1 (8-16 km)	2 (16-32 km)	3 (32-48 km)	4 (48-64 km)	5 (64-80 km)
A (N)	3.09×10^{-4}	2.79×10^{-2}	2.70×10^{-2}	8.63×10^{-3}	1.49×10^{-2}	1.19×10^{-5}	5.25×10^{-6}	2.69×10^{-6}	1.70×10^{-6}	1.22×10^{-6}
B (NNE)	5.86×10^{-5}	5.75×10^{-3}	4.71×10^{-3}	6.50×10^{-3}	1.51×10^{-2}	9.77×10^{-6}	4.35×10^{-6}	2.28×10^{-6}	1.46×10^{-6}	1.05×10^{-6}
C (NE)	1.02×10^{-5}	1.35×10^{-2}	7.03×10^{-3}	8.33×10^{-3}	1.17×10^{-2}	1.02×10^{-5}	4.57×10^{-6}	2.40×10^{-6}	1.58×10^{-6}	1.15×10^{-6}
D (ENE)	2.76×10^{-4}	1.29×10^{-2}	9.56×10^{-3}	7.43×10^{-3}	4.15×10^{-2}	1.02×10^{-5}	4.12×10^{-6}	2.13×10^{-6}	1.39×10^{-6}	1.02×10^{-6}
E (E)	1.28×10^{-3}	2.21×10^{-2}	8.91×10^{-3}	9.67×10^{-3}	3.48×10^{-3}	8.27×10^{-6}	3.27×10^{-6}	1.68×10^{-6}	1.10×10^{-6}	8.02×10^{-7}
F (ESE)	2.55×10^{-4}	4.37×10^{-3}	2.79×10^{-3}	2.56×10^{-3}	2.24×10^{-3}	7.07×10^{-6}	2.81×10^{-6}	1.45×10^{-6}	9.44×10^{-7}	6.90×10^{-7}
G (SE)	1.29×10^{-4}	1.11×10^{-3}	6.78×10^{-3}	4.54×10^{-3}	4.25×10^{-3}	4.96×10^{-6}	2.02×10^{-6}	1.04×10^{-6}	6.79×10^{-7}	4.95×10^{-7}
H (SSE)	1.61×10^{-4}	6.63×10^{-4}	6.92×10^{-4}	8.10×10^{-4}	1.12×10^{-3}	4.04×10^{-6}	1.70×10^{-6}	9.00×10^{-7}	5.97×10^{-7}	4.40×10^{-7}
I (S)	2.25×10^{-6}	5.48×10^{-4}	7.24×10^{-4}	2.69×10^{-3}	9.34×10^{-4}	2.25×10^{-6}	9.83×10^{-7}	5.44×10^{-7}	3.71×10^{-7}	2.80×10^{-7}
J (SSW)	1.29×10^{-5}	2.42×10^{-3}	2.90×10^{-3}	4.11×10^{-3}	2.12×10^{-3}	6.46×10^{-6}	2.70×10^{-6}	1.45×10^{-6}	9.82×10^{-7}	7.22×10^{-7}
K (SW)	1.87×10^{-4}	4.17×10^{-3}	5.22×10^{-3}	4.06×10^{-3}	3.02×10^{-3}	1.10×10^{-5}	4.41×10^{-6}	2.33×10^{-6}	1.56×10^{-6}	1.14×10^{-6}
L (WSW)	5.18×10^{-4}	3.87×10^{-3}	1.32×10^{-2}	2.84×10^{-3}	5.31×10^{-3}	8.64×10^{-6}	3.50×10^{-6}	1.86×10^{-6}	1.24×10^{-6}	9.13×10^{-7}
M (W)	3.43×10^{-4}	8.52×10^{-3}	1.11×10^{-2}	7.51×10^{-3}	4.62×10^{-3}	6.24×10^{-6}	2.57×10^{-6}	1.40×10^{-6}	9.40×10^{-7}	6.82×10^{-7}
N (WNW)	2.89×10^{-3}	9.16×10^{-3}	1.57×10^{-1}	4.99×10^{-2}	8.33×10^{-3}	6.43×10^{-6}	2.74×10^{-6}	1.47×10^{-6}	9.92×10^{-7}	7.22×10^{-7}
O (NW)	2.23×10^{-3}	2.08×10^{-2}	1.57×10^{-1}	3.04×10^{-2}	2.48×10^{-3}	8.22×10^{-6}	3.52×10^{-6}	1.79×10^{-6}	1.14×10^{-6}	8.21×10^{-7}
P (NNW)	3.97×10^{-3}	8.47×10^{-2}	6.28×10^{-2}	9.74×10^{-3}	6.34×10^{-3}	1.09×10^{-5}	4.70×10^{-6}	2.31×10^{-6}	1.46×10^{-6}	1.04×10^{-6}

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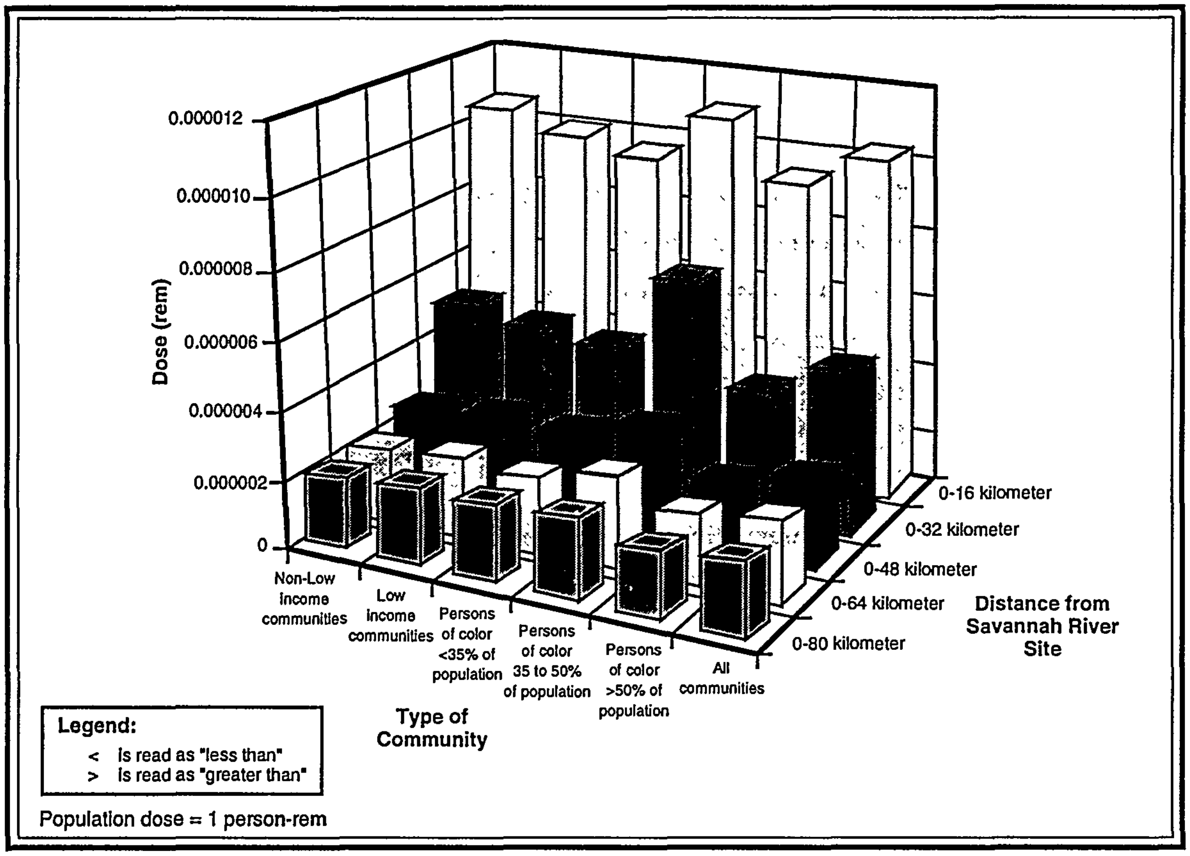
a. Sector letter is letter shown on Figure 4-1. Letters in parentheses after the sector letter indicate the compass direction of the sector.

b. km = kilometers; to convert to miles, multiply by 0.62137.



TC

Figure 4-2. Community impacts from a unit population dose.

Table 4-28. Estimated per capita 10-year dose for identified communities in 80-kilometer (50-mile) region.[a]

Distance	Low income: Less than 25 percent of population	Low income: Equal to or more than 25 percent of population	Persons of color: Less than 35 percent of population	Persons of color: 35 percent to 50 percent of population	Persons of color: Equal to or more than 50 percent of population	All communities
0-16 km[b] (0-10 miles)	0.000011	0.000010	0.000010	0.000012	0.000010	0.000011
0-32 km (0-20 miles)	0.000005	0.000005	0.000005	0.000007	0.000004	0.000005
0-48 km (0-30 miles)	0.000003	0.000003	0.000003	0.000003	0.000002	0.000003
0-64 km (0-40 miles)	0.000002	0.000002	0.000002	0.000003	0.000002	0.000002
0-80 km (0-50 miles)	0.000002	0.000002	0.000002	0.000002	0.000002	0.000002

TC

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a. Per capita dose based on a population dose of 1 person-rem. Per capita doses for other population doses can be obtained by multiplying the values in this table by the population dose.

b. km = kilometers.

The distribution of carcinogenic and criteria pollutant emissions due to routine operations, and of criteria pollutants from construction activities, would be essentially identical to those presented for airborne radiological emissions because distribution pathways would be the same. As a result, people of color or low income communities would not be disproportionately affected by nonradiological emissions from any of the alternatives. Because nonradiological pollutant emissions would have only minimal impacts for any of the alternatives, and would not be disproportionately distributed among types of communities, there are no environmental justice concerns related to these pollutants for any of the alternatives.

4.1.2 AIR RESOURCES

TC This section discusses radiological and nonradiological offsite air quality impacts from normal operation for the alternatives evaluated in this EIS. The information in this section was one of the bases for the public health effects discussed in Section 4.1.1. Appendix D includes a detailed presentation of air impacts by material category, alternative, and activities associated with each phase of the alternative.

4.1.2.1 Radiological Impacts

The radiological impact assessment indicates that the doses from total SRS airborne emissions for nuclear materials management would remain within applicable dose standards for DOE facilities. DOE conducted an assessment to establish the actions it would perform during the treatment of the materials evaluated in this EIS to facilitate its prediction of the radiological doses associated with each scenario. The assessment reviewed past and current SRS actions, identified those that are the same as or similar to potential future treatment actions, and quantified the associated airborne releases. These actions made it possible to estimate the releases associated with each material and alternative over the 10-year period of interest. The releases were converted to doses using the MAXIGASP and POPGASP computer codes (Simpkins 1994a), which calculate the dose to a hypothetical maximally exposed individual at the SRS boundary and the collective dose to the population within an 80-kilometer (50-mile) radius, respectively. Both codes utilize the GASPAR (Eckerman et al. 1980) and XOQDOQ (Sagendorf et al. 1976) modules. L11-3

The following paragraphs describe the potential variations in public dose impacts among the alternatives:

- Plutonium-242 - The Processing to Oxide and Vitrification (F-Canyon) Alternatives would result in the highest (and approximately the same) 10-year doses to the public because of the actions associated with actually converting the material to oxide or glass. The other alternatives for plutonium-242 would result in much smaller doses.

- Americium and curium - The Processing to Oxide Alternative would result in the highest 10-year dose to the public, followed by the Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative. In both cases, the operation of the separations facilities would be a major contributor of the dose, although Processing and Storage for Vitrification in the Defense Waste Processing Facility would result in a lower dose because of the shorter operating time of the separations facilities and because the vitrification activities would not occur during the 10-year period. Appendix D contains information on the annual doses that would result from vitrification.

- Neptunium - The Processing to Oxide Alternative would result in the largest 10-year dose because of the longer time required to produce the oxide. The Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would result in larger doses than the Vitrification (F-Canyon) Alternative because of the longer time needed to transfer material to the waste tanks and the larger radiological emissions from H-Area in comparison to those from F-Area. L11-3

- H-Canyon plutonium-239 solutions - The Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would result in the largest 10-year dose due to the relatively long time needed to transfer material to the waste tanks. The Processing to Oxide Alternative would yield the second highest dose because the actions would occur in H-Area (which has greater radiological emissions than F-Area); both the Vitrification (F-Canyon) and Processing to Metal Alternatives would occur in F-Area.

- H-Canyon enriched uranium solutions - The Blending Down to Low Enriched Uranium Alternative would result in the highest 10-year dose to the public, followed by the Processing to Oxide Alternative. In both cases, the conversion of material to the final product would be responsible for the larger dose due to the operation of the separations facility, as discussed in

Section 2.4. The remaining alternatives for the H-Canyon enriched uranium solutions would result in lesser doses over the 10-year period.

- Plutonium and uranium stored in vaults - The Vitrification (F-Canyon), Processing to Metal, and Processing to Oxide Alternatives would result in the same estimated doses because of the similarity of the actions associated with these alternatives. Processing and Storage for Vitrification in the Defense Waste Processing Facility would appear to result in impacts similar to those from these alternatives, but DOE could not complete the alternative involving the DWPF within the 10-year period. The other alternatives for plutonium and uranium stored in vaults would result in lesser doses.

- Mark-31 targets - The Vitrification (F-Canyon) Alternative would result in the highest dose due to the operation of the separations facilities for a relatively longer period of time in comparison to the other alternatives. The Improving Storage Alternative would be comparable to the Continuing Storage (No-Action) Alternative because a new Dry Storage Facility might not be available in the 10-year period, so activities associated with Improving Storage would be equivalent to those described for Continuing Storage.

L11-3

- Mark-16 and -22 fuels - The Blending Down to Low Enriched Uranium Alternative could result in the highest 10-year dose because of the operation of the separations facilities for a relatively longer period of time in comparison to the other alternatives. The Processing to Oxide Alternative would result in the next highest 10-year dose because of the operation of the Uranium Solidification Facility. The Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would result in a smaller dose than either of the others because the impacts of vitrification would not occur during the 10-year period. As with the Mark-31 targets, the dose from the Improving Storage Alternative would be equivalent to the dose from the Continuing Storage Alternative because a new Dry Storage Facility might not be available in the 10-year period.

- Other aluminum-clad targets - Processing and Storage for Vitrification in the Defense Waste Processing Facility is the only alternative that would involve the operation of the separations facilities and, therefore, would result in larger impacts than the other alternatives. As with the Mark-31 targets and the Mark-16 and -22 fuels, the dose from the Improving Storage Alternative would appear to be similar to the dose from the Continuing Storage Alternative because a new Dry Storage Facility might not be available in the 10-year period.

- Taiwan Research Reactor and Experimental Breeder Reactor-II fuel - The Processing to Metal, Processing to Oxide, and Vitrification (F-Canyon) Alternatives would have the greatest public health impacts and would be similar in magnitude because of the use of the same facilities for comparable durations of time. The Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would have smaller impacts because DOE would store the material for several years before transferring it to the waste tanks. The No-Action and Improving Storage Alternatives would have similar public health impacts because DOE would store the material for most or all of the 10-year period.

4.1.2.2 Nonradiological Impacts

DOE used the EPA Industrial Source Complex Short-Term No. 2 model to estimate nonradiological air pollutant concentrations. Emissions data were factored into the model along with the meteorological data discussed in Section 3.3.3. The model computed maximum boundary line concentrations at or beyond the SRS boundary. Tables 4-1 through 4-11 include the results of the modeling for pollutants of concern. Appendix D contains detailed data by material and alternative.

The following paragraphs describe potential nonradiological air impacts from the materials and their applicable alternatives. L11-3

- Stable material - The contributions to offsite concentrations of nonradiological air pollutants would be small because the material would remain in continued storage; no conversion would occur.

- Plutonium-242 - The Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would have the highest concentrations of criteria pollutants and the Processing to Metal Alternative would have the highest concentrations of gaseous fluoride and nitric acid. The higher concentrations for the Processing to Metal Alternative would be associated with the final stabilization stage. The other alternatives would result in smaller concentrations; the Continuing Storage (No-Action) Alternative would result in the smallest concentrations.

- Americium and curium - The Processing to Oxide and Vitrification (F-Canyon) Alternatives would result in the highest concentrations. The Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would result in slightly lower concentrations; the Continuing Storage Alternative would result in the smallest concentrations.

- Neptunium - The Processing to Oxide and Vitrification (F-Canyon) Alternatives would result in the highest concentrations. The Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would result in slightly lower concentrations for most pollutants. The Continuing Storage Alternative would result in the smallest concentrations.

- H-Canyon plutonium-239 solutions - The highest concentrations of carbon monoxide and sulfur dioxide would result from Processing and Storage for Vitrification in the Defense Waste Processing Facility. The highest concentrations of nitrogen oxides, gaseous fluorides, and nitric acid would be from the Processing to Metal Alternative, and would be associated with the additional conversion step. Concentrations from the Continuing Storage Alternative would be smallest.

- H-Canyon enriched uranium solutions - The Processing to Oxide and Blending Down to Low Enriched Uranium Alternatives would result in the highest concentrations for criteria pollutants. The highest concentrations for gaseous fluoride and nitric acid would result from the Blending Down to Low Enriched Uranium Alternative, and would be associated with the additional conversion step. L11-3

- Plutonium and uranium stored in vaults - The Processing to Oxide, Processing to Metal, and Vitrification (F-Canyon) Alternatives would result in the highest concentrations of carbon monoxide and sulfur dioxide. Concentrations from the Continuing Storage Alternativewould be smallest.

- Mark-31 targets - The highest concentrations would result from the Vitrification (F-Canyon) Alternative. The other alternatives would result in slightly lower concentrations.

- Mark-16 and -22 fuels - The highest concentrations of carbon monoxide and sulfur dioxide would result from the Processing to Oxide and Blending Down to Low Enriched Uranium Alternatives. The highest concentrations of nitrogen oxides, gaseous fluorides, and nitric acid would result from the Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative. The highest concentrations would be associated with the conversion step.

- Other aluminum-clad targets - The Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would be the only alternative with additional concentrations of nonradiological air pollutants. L11-3

- Taiwan Research Reactor and Experimental Breeder Reactor-II fuel - The highest concentrations would result from the Vitrification (F-Canyon) Alternative. The other alternatives would result in slightly lower concentrations.

Virtually all nonradiological air pollutant emissions for each material are associated with activities in F- and H-Areas. These emissions can be attributed to the F- and H-Area main stacks, diesel generators, and storage tanks. Emissions from the generators and storage tanks do not vary by material or treatment alternative, and thus are part of the facility baseline. These emissions, which are accounted for in Section 3.3.3, are not included in the incremental modeling results listed in Tables 4-1 through 4-11. Table 4-29 lists regulatory standards for the applicable pollutants, which can be compared directly to the sum of the concentrations for each pollutant that comprises the existing ambient concentration (listed in Table 3-5) and the incremental increase (listed in Tables 4-1 through 4-11). Total concentrations attributable to any alternative would be much less than the regulatory standards.

Table 4-29. Regulatory standards for nonradiological air pollutants (micrograms per cubic meter).[a] L11-19

Pollutant	Averaging time	Regulatory standard
Carbon monoxide	1 hour	40,000
	8 hours	10,000
Nitrogen oxides	Annual	100
Sulfur dioxide	3 hours	1,300
	24 hours	365
	Annual	80
Gaseous fluorides	12 hours	3.7
(as HF)	24 hours	2.9
	1 week	1.6
	1 month	0.8
Nitric acid	24 hours	125
	Annual	None

a. Source: SCDHEC (1976).

4.1.3 WATER RESOURCES

TC This section describes the normal effects associated with the alternatives. This information was one of the bases for the health effects discussed in Section 4.1.1. Based on the data in Tables 4-1 through 4-11, DOE expects minimal impacts to either surface water or groundwater.

Because normal operations would not involve releases to groundwater, DOE has limited this section to surface-water impacts. The major sources of liquid effluents from involved facilities would be process cooling water and steam condensate that could contain small quantities of radionuclides and chemicals. The exposure pathways considered are drinking water, fish ingestion, shoreline exposure, swimming, and boating. Usage factors for the maximally exposed individual are consistent with regularly published SRS environmental reports (e.g., Arnett, Karapatakis, and Mamatey 1994). As described below, DOE used a mathematical model to calculate the dose to the maximally exposed offsite individual and the collective dose to the offsite population.

DOE conducted an assessment to establish the actions it would perform during the treatment of the materials evaluated in this EIS. The assessment reviewed past and current actions at the SRS, identified those that are the same or similar to future alternatives, and quantified the associated liquid releases; this made it possible to estimate the releases associated with each material and alternative over the 10-year period of interest.

4.1.3.1 Radiological Impacts

TE Calculations of radiological doses through water pathways based on these releases are supported by the use of LADTAP II, a computer code developed by the U.S. Nuclear Regulatory Commission (NRC) to estimate radiation doses associated with normal reactor system liquid effluent releases to individuals, populations groups, and biota. LADTAP II uses the models in the NRC Regulatory Guide 1.109 (NRC 1977) to calculate doses received from water and fish ingestion and from recreational water activities.

L11-3 Any radionuclide releases to surface water resulting from the alternatives would be to SRS streams that discharge to the Savannah River. For all alternatives, the ingestion of fish containing cesium-137 would contribute most of the exposure to both the maximally exposed individual and the population. Plutonium and uranium isotopes ingested with drinking water would be smaller contributors. Tables 4-1 through 4-11 list both the maximally exposed individual dose and the population dose due to liquid releases, and Appendix D contains detailed information by phases required to implement each alternative.

The variations between the predicted impacts from the stabilization alternatives on water resources would result from the same influences as those listed in Section 4.1.2.1 for radiological impacts to air resources. The variations would be due to facility operation, not to an exposure pathway. L11-3

4.1.3.2 Nonradiological Impacts TE

This assessment also compared chemical releases with applicable water quality standards. These standards are based on the preservation of aquatic biota populations, human health, and aesthetics (i.e., taste and odor). Figure 3-5 shows that none of the stabilization actions would occur within the 100-year floodplain. DOE would treat sanitary waste associated with personnel necessary to perform the selected treatment alternatives in existing sewage treatment plants; discharges from these plants (e.g., to L-Lake from L-Area, to Fourmile Branch from F-Area, to Fourmile Branch from H-Area) would continue to meet National Pollutant Discharge Elimination System permit limits.

Under the analyzed alternatives, process cooling water treatment would result in releases of the following concentrations from F-Area to Upper Three Runs Creek: TC

- Nitrate - 40 micrograms per liter
- Ammonia - 30 micrograms per liter
- Manganese - 10 micrograms per liter
- Uranium - 20 micrograms per liter
- Lead - 6 micrograms per liter
- Nickel - 50 micrograms per liter
- Chromium - 20 micrograms per liter
- Aluminum - 200 micrograms per liter
- Copper - 10 micrograms per liter
- Zinc - 70 micrograms per liter

Similar or lower concentrations would be released from H-Area,with the exception of those for nitrate and ammonia, which would be 100 and 500 micrograms per liter, respectively. Although proposed or final Federal drinking water standards do not apply to discharges, the SRS discharge concentrations would not exceed these standards (Arnett, Karapatakis, and Mamatey 1993). The discharges would also comply with South Carolina Water Quality Standards (SC 1994). In general, the release concentrations would be no greater than those measured in Upper Three Runs Creek and Fourmile Branch (Arnett 1993, 1994), with the exception of zinc and ammonia; however, zinc concentrations in the discharge would be only a small fraction of the South Carolina Water Quality Standards, which are based on the taste and odor of drinking water. Ammonia concentrations in the TE TC discharge (of which only H-Area releases would exceed stream concentrations) would be well within state standards. Lead, nickel, chromium, and copper were generally not detected in Upper Three Runs Creek and Fourmile Branch in 1993. The release concentrations of these metals would be no greater than those measured in 1992 and are well within state standards.

4.1.4 UTILITIES

DOE based its estimates of water, electricity, steam, and fuel annual consumption rates on past operational experience and the projected usage for each material and alternative. Tables 4-1 through 4-11 list utility information for the alternatives. Appendix D presents annual impacts for the various phases of stabilization by material. As with other environmental factors described in this chapter, the No-Action Alternative for a given material generally would result in lower estimated utility consumption over the 10-year period of interest.

L11-3 The following paragraphs describe differences in utilities consumption by alternative.

- Stable material - The utilities consumption associated with this material would be the continuation of current storage activities under the Continuing Storage Alternative.

- Plutonium-242 - The Processing to Oxide and Vitrification (F-Canyon) Alternatives would have the lowest electricity, steam, and fuel usage of the alternatives; the Continuing Storage Alternative would have the greatest. The Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would have the lowest water usage and the Processing to Metal the greatest.

- Americium and curium - The Continuing Storage Alternative would have the highest level of utilities usage, twice the level of current utilities consumption. The Vitrification (F-Canyon) Alternative would have the lowest level of utilities consumption.

- Neptunium - The Vitrification (F-Canyon) Alternative would have the greatest level of electricity, steam, and fuel usage. The Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would have a somewhat lower level of utilities usage than the other alternatives, but the impacts would be understated because vitrification would not occur during the 10-year period of interest. Processing to Oxide would have the greatest water usage and Processing and Storage for Vitrification in the DWPF would have the smallest.

- H-Canyon plutonium-239 - The Processing to Oxide Alternative would have the lowest electricity, steam, and fuel usage and the Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would have the highest usage. Water usage would be lowest for the Continuing Storage Alternative and highest for the Processing to Metal Alternative.

- H-Canyon enriched uranium solutions - The Processing to Oxide and Blending Down to Low Enriched Uranium Alternatives would have the lowest utility usage of the alternatives, a fourth to a third of the usage predicted for the Processing and Storage for Vitrification in the Defense Waste Processing Facility and Continuing Storage Alternatives.

- Plutonium and uranium stored in vaults - The Continuing Storage Alternative would have the highest water usage. The Improving Storage Alternative would have the lowest utilities usage. Other alternatives would have similar utilities usage due to similarity of operations and operation of the same facilities.

- Mark-31 targets - The Continuing Storage and Improving Storage Alternatives would have low utilities consumption, much less than the other alternatives. The Vitrification (F-Canyon) Alternative would have the highest utilities usage.

TC

- Mark-16 and -22 fuels - Water and electricity usage for the Processing and Storage for Vitrification at the Defense Waste Processing Facility would be somewhat higher than for the other five alternatives. Processing to Oxide, Blending Down to Low Enriched Uranium, and Processing and Storage for Vitrification in the Defense Waste Processing Facility would be similar in their levels of steam and fuel usage. The Continuing Storage and Improving Storage (traditional schedule and accelerated schedule) Alternatives would have the lowest utilities consumption, much less than the other alternatives.

- Other aluminum-clad targets - The Processing and Storage for Vitrification in the Defense Waste Processing Facility and Improving Storage (accelerated schedule) Alternatives would have relatively low usage levels for all utilities. The Continuing Storage and Improving Storage Alternatives would consume only electricity.

- Taiwan Research Reactor and Experimental Breeder Reactor-II fuel - The Processing to Metal, Processing to Oxide, and Vitrification (F-Canyon) Alternatives would have the greatest utility usage and would be similar in magnitude because of the use of the same facilities for comparable durations of time. The Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would have smaller impacts because DOE would store the material for several years before transferring it to the waste tanks, thus using less utilities. The No-Action and Improving Storage Alternatives would have similar utilities consumption because DOE would store the material for most or all of the 10-year period. TC

4.1.5 WASTE MANAGEMENT

Tables 4-1 through 4-11 list waste generation information for the alternatives; the estimates in these tables are based on current and past SRS operations (WSRC 1995a). The SRS generates several different types of waste, including low-level waste, high-level waste, transuranic waste, and mixed waste. Low-level waste constitutes a substantial portion of the generated waste and typically contains relatively small amounts of dispersed radioactive material. Compaction is often employed to reduce the volume of this type of waste and to minimize disposal space. High-level waste at the SRS is a liquid resulting from processing operations in the canyon facilities; DOE will treat this waste at the TC Defense Waste Processing Facility and convert it to a solid glass material encapsulated in stainless-steel canisters. This EIS expresses the generation of high-level waste as both the volume of high-level liquid waste and "equivalent DWPF canisters," even though this facility will not produce canisters during the early portion of the 10-year time period covered by this EIS. The volumes of liquid waste reported in this section are the volumes as they leave the canyon, and do not reflect final volumes that would enter the waste tanks after concentration and evaporation. The use of equivalent DWPF canisters for measuring high-level waste provides a better comparison among alternatives because liquid waste can be diluted or concentrated such that the volume of liquid is not an accurate indicator of the actual waste content.

In general, alternatives that would involve processing activities would tend to result in higher estimates of high-level waste, saltstone, and DWPF canisters in comparison to nonprocessing alternatives. However, alternatives that would involve processing activities would tend to result in comparable or L11-3 lower estimates of low-level waste over the 10-year period in comparison to the Continuing Storage Alternative or nonprocessing alternatives.

The following paragraphs describe differences in waste generation by alternative. To avoid listing each waste type repeatedly, these paragraphs discuss high-level waste with the understanding that the discussion applies equally to equivalent DWPF canisters and saltstone. Similarly, transuranic waste and hazardous/mixed waste would follow comparable (but not identical) trends; therefore, the paragraphs discuss only transuranic waste to avoid confusion. Low-level waste is discussed separately.

- Plutonium-242

  High-Level Waste: The Continuing Storage and Processing to Metal Alternatives would generate comparable volumes of high-level waste due to the storage of material in solution form. The Processing to Metal Alternative would generate more high-level waste during processing activities but less after the completion of stabilization. Thus, the total would be comparable to that of Continuing Storage. The Processing to Oxide and Vitrification (F-Canyon) Alternatives would generate similar quantities of high-level waste because of the similarity of activities and durations involved in stabilizing the material. These volumes would be less than those from Processing to Metal because DOE could complete the actions earlier in the 10-year period, so the smaller waste generation after stabilization becomes important.

  Transuranic Waste: FB-Line and HB-Line would generate transuranic waste while in operation. Therefore, Processing to Metal, Processing to Oxide, and Vitrification (F-Canyon) would all generate transuranic waste because they would use these facilities. The durations for active steps in the Processing to Oxide and Vitrification (F-Canyon) Alternatives would be comparable, so these two alternatives would generate similar waste volumes. L11-3

  Low-Level Waste: The volumes of low-level waste generally would be similar across the alternatives. However, the Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would result in the generation of smaller amounts of waste during the 10-year period because additional waste generation from the vitrification would not occur.

- Americium and curium

  High-Level Waste: The Processing to Oxide Alternative would result in the greatest generation of high-level waste because of the relatively long time that would be required to convert the material to an oxide compared to the other alternatives, as discussed in Section 2.4. The Vitrification (F-Canyon) and the Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternatives would generate larger volumes of high-level waste compared to those from Continuing Storage because of the use of the separations facilities.

Transuranic Waste: No alternatives would use the traditional FB-Line or HB-Line [although a modified portion of F-Canyon would be used for the Vitrification (F-Canyon) Alternative]; therefore, no transuranic waste would be generated.

Low-Level Waste: The Processing to Oxide Alternative, because of the relatively long time that would be required to convert the material to an oxide, would generate larger volumes of low-level waste than the other alternatives. The other alternatives would generate similar volumes of low-level waste.

- Neptunium

L11-3

High-Level Waste: The Processing to Oxide Alternative would result in the greatest generation of high-level waste because of the relatively long time that would be required to convert the material to an oxide in comparison to the other alternatives. The other alternatives would generate similar waste volumes because of the similarity in activities and durations associated with the activities. The completion of stabilization before the end of the 10-year period would result in lower post-stabilization generation of waste, so the Continuing Storage Alternative (which would generate waste for the full 10 years) would generate a similar volume of waste.

Transuranic Waste: Because only the Processing to Oxide Alternative would involve the active operation of either the FB- or HB-Line, it is the only alternative that would generate transuranic waste.

Low-Level Waste: The Processing to Oxide Alternative, because of the relatively long time that would be required to convert the material to an oxide, would generate larger volumes of low-level waste than the other alternatives. The other alternatives would generate similar volumes of low-level waste over the 10-year period.

- H-Canyon plutonium-239 solutions

High-Level Waste: The Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would generate the greatest amount of high-level waste because of the longer operating time of the separations facilities to transfer the material to the high-level waste tanks. The Processing to Oxide Alternative would generate the least amount of high-level waste because of the relatively short time that would be required to achieve stabilization.

Transuranic Waste: Only the Processing to Metal and Processing to Oxide Alternatives would involve active operation of either the FB- or HB-Line; therefore, these are the only alternatives that would generate transuranic waste. The Processing to Oxide Alternative would generate slightly greater volumes because the waste generation rate for H-Area is greater.

Low-Level Waste: Most of the alternatives would generate similar volumes of low-level waste; however, the Vitrification (F-Canyon) Alternative would generate smaller volumes because it would complete the stabilization activities sooner. Even though the 10-year totals suggest that the Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would generate similar volumes compared to other alternatives, that alternative would generate additional volumes at the time of vitrification, which would occur after 10-year period.

- H-Canyon enriched uranium solutions L11-3

High-Level Waste: The Continuing Storage Alternative would generate the greatest volume of high-level waste because of the higher generation rates associated with continued management of the solutions. The Processing to Oxide Alternative would generate less waste because of the shorter operating time for H-Canyon and the minimal generation of waste associated with the Uranium Solidification Facility.

Transuranic Waste: None of the alternatives would involve the operation of either the HB- or FB-Line; therefore, no transuranic waste would be generated.

Low-Level Waste: The Processing to Oxide and Blending Down to Low Enriched Uranium Alternatives would generate the least amount of low-level waste because DOE could complete them sooner. In contrast, the Processing and Storage for Vitrification in the Defense Waste Processing Facility and Continuing Storage Alternatives would generate steady amounts of low-level waste for the entire 10-year period, and Processing and Storage for Vitrification in the Defense Waste Processing Facilities probably would generate additional waste at the time of vitrification.

- Plutonium and uranium stored in vaults

High-Level Waste: The Continuing Storage and Improving Storage Alternatives would result in no high-level waste. Waste generation for Processing to Oxide, Processing to Metal, Vitrification (F-Canyon), and Processing and Storage for Vitrification in the Defense Waste Processing Facility would be similar due to their similar activities, facilities, and durations.

Transuranic Waste: The waste generation estimates would be similar across the alternatives, although the Continuing Storage Alternative would result in a generally lower estimate than the other alternatives. Waste generation for Processing to Oxide, Processing to Metal, Vitrification (F-Canyon), and Processing and Storage for Vitrification in the Defense Waste Processing Facility would be similar due to their similar activities, facilities, and durations.

Low-Level Waste: Low-level waste generation estimates would be similar across the alternatives, although additional waste could be generated during vitrification activities as part of the Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative. Waste generation for Processing to Oxide, Processing to Metal, Vitrification (F-Canyon), and Processing and Storage for Vitrification in the Defense Waste Processing Facility would be similar due to their similar activities, facilities, and durations. L11-3

- Mark-31 targets

High-Level Waste: The Continuing Storage and Improving Storage Alternatives would generate similar volumes of waste because a new Dry Storage Facility might not be available until near the end of the 10-year period. Because construction would be completed sooner, the Improving Storage (Accelerated Construction Schedule) Alternative would result in fewer impacts. The Processing to Metal, Processing to Oxide, and Vitrification (F-Canyon) Alternatives would generate similar waste volumes because of the use of similar facilities for comparable durations. The Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would generate the most high-level waste because of the longer operating time of the separations facilities to transfer the material to the high-level waste tanks.

Transuranic Waste: All alternatives would generate hazardous and mixed waste because of the generation of this type of waste from the reactor basins. In addition, alternatives that would involve the operation of FB-Line also would generate transuranic waste, and the volume generated would not vary dramatically among the alternatives.

Low-Level Waste: The Continuing Storage and Improving Storage Alternatives could generate the greatest volumes of low-level waste because of the higher generation rates associated with continued storage of the material, consistent with historic basin operations. Alternatives that would involve the operation of the separations facilities would result in lower amounts of low-level waste because DOE would stabilize the material before the end of the 10-year period. However, under the Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative, the material would not be in a stabilized form at the end of the 10-year period, resulting in the generation of additional low-level waste after the 10-year period.

- Mark-16 and -22 fuels:

High-Level Waste: The Continuing Storage and Improving Storage Alternatives would generate similar volumes of waste because a new Dry Storage Facility might not be available until near the end of the 10-year period. The other alternatives would generate similar waste volumes because of the facilities used and similar durations for actions. L11-3

Transuranic Waste: Storing the fuels in the reactor basins for any period of time would result in the generation of some hazardous and mixed waste in addition to that associated with storage of such material in the reactor basin. Waste volumes would be similar among the alternatives. No transuranic waste would be generated under any of the alternatives.

Low-Level Waste: Processing and Storage for Vitrification in the Defense Waste Processing Facility could result in the greatest volume of waste due to greater low-level waste generation associated with the transfer of material to the waste tanks. The other alternatives for this material would result in volumes similar to each other because they would use similar processes in facilities for similar durations.

- Other aluminum-clad targets

High-Level Waste: The Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would result in the greatest volume of high-level waste due to the operation of the separations facilities. The Continuing Storage and Improving Storage

Alternatives would generate similar volumes of waste because a new Dry Storage Facility might not be available until near the end of the 10-year period.

Transuranic Waste: Hazardous/mixed waste is associated with the storage of materials in the reactor basin. The Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would result in less waste because the material would be removed from the reactor basin before the end of the 10-year period. L11-3

Low-Level Waste: The Continuing Storage and Improving Storage Alternatives would generate similar volumes of waste because a new Dry Storage Facility might not be available until near the end of the 10-year period. The Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would result in a lower 10-year generation of low-level waste compared to other alternatives; however, additional waste would probably be generated during vitrification.

- Taiwan Research Reactor and Experimental Breeder Reactor-II fuel

High-level waste: The Continuing Storage and Improving Storage Alternatives would generate similar volumes of waste because a new Dry Storage Facility could not be available until near the end of the 10-year period. The Improving Storage (Accelerated Construction Schedule) Alternative would result in fewer impacts because construction would be completed sooner. The Processing to Metal, Processing to Oxide, and Vitrification (F-Canyon) Alternatives would generate similar waste volumes because of the use of similar facilities for comparable durations. The Processing and Storage for Vitrification in the Defense Waste Processing Facility Alternative would generate the most DWPF canisters because of the relatively high plutonium content of the liquid waste, which DOE would have dilute in many canisters to alleviate criticality concerns. TC

Transuranic waste: Only alternatives that would involve the operation of a portion of the FB-Line [i.e., Processing to Metal, Processing to Oxide, and Vitrification (F-Canyon)] would generate transuranic and hazardous or mixed waste. Because of the similarity of operations and the use of the same facility for all three alternatives, however, the amount of generated waste would not vary greatly among these alternatives.

Low-level waste: Continuing Storage, Improving Storage, and Improving Storage (Accelerated Construction Schedule) would generate similar volumes of low-level waste. The

Processing and Storage for Vitrification in the Defense Waste Processing Facility would generate a waste volume similar to that from Continuing Storage because DOE would continue to store the material for most of the 10-year period as it would for the Continuing Storage Alternative. The Processing to Metal, Processing to Oxide, and Vitrification (F-Canyon) Alternatives would generate similar volumes of low-level waste because of the use of the same facilities for comparable durations of time.

4.1.6 TRAFFIC

DOE analyzed impacts from each alternative to workers and members of the public from traffic activities. Road traffic related to facility operations would remain at or below current SRS levels because none of the alternatives would require the addition of employees to the SRS workforce. Rail traffic could increase slightly due to the movement of spent fuel (HNUS 1994b).

4.2 Impacts from Accidents

This section summarizes risks to members of the public and workers from potential facility or transportation accidents associated with the alternatives for management of the nuclear materials stored at the SRS. This EIS defines an accident as a series of unexpected or undesirable events leading to a release of radioactive or hazardous material within a facility or to the environment. All the alternatives discussed in this EIS (including No Action) have a potential for accidents from both a facility and a transportation perspective. TC

4.2.1 IMPACTS FROM FACILITY ACCIDENTS

DOE performs safety analyses for the SRS facilities that process and store nuclear materials to identify and describe potential accidents. The information from these analyses, along with information on inventories of hazardous chemicals or radioactive materials involved with each alternative, provides estimates of potential impacts from such accidents. Some of the alternatives for some of the materials discussed in this EIS would involve new facilities, different material forms, or different chemical compositions that the existing safety analysis reports have not analyzed. DOE believes that analyses for existing materials and inventories would be representative of accident impacts. If technical studies for existing facilities indicate that the safety analyses would not account for new uses or new material compositions, DOE would perform additional safety analyses, as required by applicable DOE Orders.

TC The accidents analyzed could be the result of external events (aircraft crashes, nearby explosions), internal events (equipment failures, human errors), or natural phenomena (earthquakes, tornadoes). The analysis considered a spectrum of accidents (i.e., high- and low-frequency events and large- and small-consequence events) that could result in the release of radioactive and/or hazardous materials. For radiological accidents, this section presents consequences in terms of the dose to an individual or the collective dose to a population. DOE has converted these potential doses to health effects in the form of latent cancer fatalities. For hazardous material releases, consequences are presented as chemical concentrations.

To estimate the doses that would result from radiological accidents, DOE established an initial baseline by assuming a release of 1 curie of each type of radionuclide from a point on the SRS that is representative of the location of the nuclear facilities. Mathematical models predicted the dose to an individual hypothetically located 640 meters (2,100 feet) from the point of release. The mathematical models account for such factors as the meteorological conditions at the time of the accident and the rate at which the accident would deposit radioactive material over the landscape (i.e., deposition rate). DOE used the distance of 640 meters (2,100 feet) to estimate the impacts to an uninvolved worker (i.e., a worker not in the immediate vicinity of an accident, but potentially in a nearby facility or work area that is directly in the path of a radioactive plume). Similarly, DOE used the model to estimate the dose to an individual hypothetically located at a point on the SRS boundary that is directly in the path of a radioactive plume; this simulates potential impacts to a maximally exposed member of the public. DOE calculated the collective dose to the offsite population for individuals living within 80 kilometers (50 miles) of the Site who would be in the path of any release plume.

TC After developing the baseline information, DOE used the estimated amount of radioactive material released during each hypothetical accident to calculate potential corresponding doses to an uninvolved worker, the maximally exposed offsite individual, and the offsite population. The calculation of the estimated number of latent cancer fatalities used the conversion factors of 0.0005 latent cancer fatality per person-rem and 0.0004 latent cancer fatality per person-rem (0.0008 for projected individual doses above 20 rem) to determine health effects to the public and for uninvolved workers, respectively, as discussed in Section 4.1.1.1.

DOE multiplied the resulting accident consequences, in terms of latent cancer fatalities, by the estimated accident frequency to calculate the point estimate of accident risk. The annualized point estimate of risk is provided to enable the consideration of accidents that might not have the highest consequence but that might pose a greater risk due to a higher frequency.

An example of this concept is the Continuing Storage Alternative accidents related to the H-Canyon plutonium solutions listed in Table E-7. The inadvertent transfer from a processing vessel to the ground outside the H-Canyon building would result in the greatest consequence: 4.1 latent cancer fatalities per occurrence (Note: this number is in bold type in Table E-7). Because this accident is likely to occur only once in every 2,500 years [Table E-7 lists this frequency as 4.00E-04 (0.0004)], a time-weighted average of these consequences over the accident frequency time span (i.e., consequence times frequency) would result in an annualized point estimate of risk of 0.0017 latent cancer fatality per year. Although an unpropagated fire in a solution vessel would produce lower consequences – 1.3 latent cancer fatalities per occurrence – DOE estimates that this accident would occur once in every 45 years (a frequency of 0.0202), resulting in a higher point estimate of risk (0.026 latent cancer fatality per year). By factoring in the accident probability, DOE can compare the resulting risks, although the health effects (i.e., latent cancer fatalities) would not be realized if an accident did not occur. The frequency, consequence, and risk values for accident analysis would not depend on the durations of the phases of an alternative; this means that if DOE put a material in storage earlier or later than the estimated durations in Chapter 2, the accident impacts would be the same.

TE

TC

TC

This analysis discusses potential accidental radiological impacts to involved workers qualitatively; however, in the event of a criticality, the result could be prompt fatalities. For personnel other than workers who would be nearby, the impact would be delayed. The human health effect of concern is the delayed development of cancer (latent cancer) that proves fatal.

TE

Tables 4-30 and 4-31 summarize the projected impacts of accidents on the population, maximally exposed offsite individual, and uninvolved worker. These tables list the alternatives for each material group. To facilitate comparison among the alternatives, the tables list two parameters (i.e., Table 4-30 deals with latent cancer fatalities and Table 4-31 deals with point estimate of risk) for each material group. In addition, the tables list only the maximum values from the various phases of an alternative for each receptor group. For each material, the tables contain a reference to a more detailed table in Appendix E. Actions such as characterizing materials and other monitoring are represented by accident analyses for the Continuing Storage (No-Action) Alternative for each material group. Existing storage of material is part of each No-Action Alternative.

L11-3
TC

Table 4-30 lists the estimated increases in latent cancer fatalities resulting from the calculated population dose of the maximum consequence accident. This projected increase in latent cancer fatalities is conservative and could result only if the postulated, yet highly unlikely, accident occurred during highly unfavorable meteorological conditions. The table lists the potential population impacts

Table 4-30. Estimated impacts from maximum projected consequence accidents (latent cancer fatalities) for each alternative.

	Alternative						
Material/Receptor Group	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Vitrification (F-Canyon)	Processing and Storage for Vitrification (DWPF)	Improving Storage
Plutonium-242 (Table E-4)							
Population	6.5	6.5	6.5	NA	6.5	38	NA
Maximally exposed individual	8.8×10^{-4}	8.8×10^{-4}	8.8×10^{-4}		8.8×10^{-4}	3.4×10^{-3}	
Uninvolved worker	6.6×10^{-3}	6.6×10^{-3}	6.6×10^{-3}		6.6×10^{-3}	(a)	
Americium and curium (Table E-5)							
Population	3.1	NA	6.5	NA	6.5	38	NA
Maximally exposed individual	5.2×10^{-4}		8.8×10^{-4}		8.8×10^{-4}	3.4×10^{-3}	
Uninvolved worker	1.8×10^{-2}		6.6×10^{-3}		6.6×10^{-3}	(a)	
Neptunium (Table E-6)							
Population	4.1	NA	4.1	NA	6.5	38	NA
Maximally exposed individual	7.1×10^{-4}		7.1×10^{-4}		8.8×10^{-4}	3.4×10^{-3}	
Uninvolved worker	2.4×10^{-2}		2.4×10^{-2}		6.6×10^{-3}	(a)	
H-Canyon plutonium-239 solutions (Table E-7)							
Population	4.1	6.5	4.1	NA	6.5	38	NA
Maximally exposed individual	7.1×10^{-4}	8.8×10^{-4}	7.1×10^{-4}		8.8×10^{-4}	3.4×10^{-3}	
Uninvolved worker	2.4×10^{-2}	6.6×10^{-3}	2.4×10^{-2}		6.6×10^{-3}	(a)	
H-Canyon enriched uranium solutions (Table E-8)							
Population	0.14	NA	0.14	0.14	NA	38	NA
Maximally exposed individual	3.4×10^{-4}		3.4×10^{-4}	3.4×10^{-4}		3.4×10^{-3}	
Uninvolved worker	1.2×10^{-5}		6.6×10^{-3}	6.6×10^{-3}		(a)	
Plutonium and uranium stored in vaults (Table E-9)							
Population	0.31	4.1	4.5	NA	4.1	38	0.62
Maximally exposed individual	5.1×10^{-5}	7.1×10^{-4}	5.5×10^{-4}		7.1×10^{-4}	3.4×10^{-3}	1.0×10^{-4}
Uninvolved worker	9.1×10^{-4}	2.4×10^{-2}	1.8×10^{-3}		2.4×10^{-2}	(a)	1.8×10^{-3}
Mark-31 targets (Table E-10)							
Population	8.9×10^{-3}	6.5	6.5	NA	6.5	38	8.9×10^{-3}
Maximally exposed individual	2.7×10^{-6}	8.8×10^{-4}	8.8×10^{-4}		8.8×10^{-4}	3.4×10^{-3}	2.7×10^{-6}
Uninvolved worker	3.1×10^{-5}	6.6×10^{-3}	6.6×10^{-3}		6.6×10^{-3}	(a)	3.1×10^{-5}

L11-3
TC

Table 4-30. (continued).

Material/Receptor Group	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Vitrification (F-Canyon)	Processing and Storage for Vitrification (DWPF)	Improving Storage
Mark-16 and -22 fuels (Table E-11)							
Population	8.9×10^{-3}	NA	4.1	4.1	NA	38	8.9×10^{-3}(b)
Maximally exposed individual	2.7×10^{-6}		7.1×10^{-4}	7.1×10^{-4}		3.4×10^{-3}	2.7×10^{-6}(b)
Uninvolved worker	3.1×10^{-5}		2.4×10^{-2}	2.4×10^{-2}		(a)	3.1×10^{-5}(b)
Other aluminum-clad targets (Table E-3)							
Population	8.9×10^{-3}	NA	NA	NA	NA	38	8.9×10^{-3}(b)
Maximally exposed individual	2.7×10^{-6}					3.4×10^{-3}	2.7×10^{-6}(b)
Uninvolved worker	3.1×10^{-5}					(a)	3.1×10^{-5}(b)
TRR fuel and EBR-II slugs[c] (Table E-12)							
Population	(d)	6.5	6.5	NA	6.5	38	3.5×10^{-6}
Maximally exposed individual	2.1×10^{-6}	8.8×10^{-4}	8.8×10^{-4}		8.8×10^{-4}	3.4×10^{-3}	2.1×10^{-6}
Uninvolved worker	4.7×10^{-4}	6.6×10^{-3}	6.6×10^{-3}		6.6×10^{-3}	(a)	4.7×10^{-4}

L11-3
TC

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a. The number of latent cancer fatalities is not calculated for this receptor group because the dose (4,000 rem) would result in death within a few days. This accident scenario assumes that the worker is 100 meters (109 yards) from the release.

b. Accident impacts for dry storage and accelerated dry storage would be the same. Table E-11 lists values that represent both dry storage and accelerated dry storage.

c. TRR = Taiwan Research Reactor; EBR = Experimental Breeder Reactor.

d. These data are not available.

Table 4-31. Estimated impacts from maximum projected risk accidents (latent cancer fatalities per year) for each alternative.

	Alternative						
Material/Receptor Group	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Vitrification (F-Canyon)	Processing and Storage for Vitrification (DWPF)	Improving Storage
Plutonium-242 (Table E-4)							
Population	8.8×10^{-2}	8.8×10^{-2}	8.8×10^{-2}	NA	8.8×10^{-2}	8.8×10^{-2}	NA
Maximally exposed individual	1.5×10^{-5}	1.5×10^{-5}	1.5×10^{-5}		1.5×10^{-5}	1.5×10^{-5}	
Uninvolved worker	2.6×10^{-4}	2.6×10^{-4}	2.6×10^{-4}		2.6×10^{-4}	2.6×10^{-4}	
Americium and curium (Table E-5)							
Population	4.3×10^{-4}	NA	8.8×10^{-2}	NA	8.8×10^{-2}	8.8×10^{-2}	NA
Maximally exposed individual	5.7×10^{-8}		1.5×10^{-5}		1.5×10^{-5}	1.5×10^{-5}	
Uninvolved worker	1.6×10^{-6}		2.6×10^{-4}		2.6×10^{-4}	2.6×10^{-4}	
Neptunium (Table E-6)							
Population	2.6×10^{-2}	NA	2.6×10^{-2}	NA	8.8×10^{-2}	2.6×10^{-2}	NA
Maximally exposed individual	3.6×10^{-6}		3.6×10^{-6}		1.5×10^{-5}	3.6×10^{-6}	
Uninvolved worker	1.7×10^{-5}		1.7×10^{-5}		2.6×10^{-4}	1.2×10^{-4}	
H-Canyon plutonium-239 solutions (Table E-7)							
Population	2.6×10^{-2}	8.8×10^{-2}	2.6×10^{-2}	NA	8.8×10^{-2}	2.6×10^{-2}	NA
Maximally exposed individual	3.6×10^{-6}	1.5×10^{-5}	3.6×10^{-6}		1.5×10^{-5}	3.6×10^{-6}	
Uninvolved worker	1.7×10^{-5}	2.6×10^{-4}	1.7×10^{-5}		2.6×10^{-4}	1.2×10^{-4}	
H-Canyon enriched uranium solutions (Table E-8)							
Population	1.2×10^{-3}	NA	1.2×10^{-3}	1.2×10^{-3}	NA	2.6×10^{-2}	
Maximally exposed individual	9.6×10^{-7}		9.6×10^{-7}	9.6×10^{-7}		3.6×10^{-6}	
Uninvolved worker	1.3×10^{-7}		1.5×10^{-6}	1.5×10^{-6}		1.2×10^{-4}	
Plutonium and uranium stored in vaults (Table E-9)							
Population	6.1×10^{-5}	2.6×10^{-2}	5.7×10^{-2}	NA	2.6×10^{-2}	6.5×10^{-3}	1.2×10^{-4}
Maximally exposed individual	1.0×10^{-8}	3.6×10^{-6}	2.2×10^{-6}		3.6×10^{-6}	8.9×10^{-7}	2.0×10^{-8}
Uninvolved worker	1.8×10^{-7}	1.7×10^{-5}	1.9×10^{-6}		1.7×10^{-5}	4.3×10^{-6}	3.6×10^{-7}
Mark-31 targets (Table E-10)							
Population	3.7×10^{-6}	8.8×10^{-2}	8.8×10^{-2}	NA	8.8×10^{-2}	8.8×10^{-2}	3.7×10^{-6}
Maximally exposed individual	4.9×10^{-9}	1.5×10^{-5}	1.5×10^{-5}		1.5×10^{-5}	1.5×10^{-5}	4.9×10^{-9}
Uninvolved worker	6.1×10^{-9}	2.6×10^{-4}	2.6×10^{-4}		2.6×10^{-4}	2.6×10^{-4}	6.1×10^{-9}

L11-3
TC

Table 4-31. (continued).

Material/Receptor Group	Alternative Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Vitrification (F-Canyon)	Processing and Storage for Vitrification (DWPF)	Improving Storage
Mark-16 and -22 fuels (Table E-11)							
Population	3.7×10^{-6}	NA	2.6×10^{-2}	2.6×10^{-2}	NA	2.6×10^{-2}	3.7×10^{-6}
Maximally exposed individual	4.9×10^{-9}		3.6×10^{-6}	3.6×10^{-6}		3.6×10^{-6}	4.9×10^{-9}
Uninvolved worker	6.1×10^{-9}		1.7×10^{-5}	1.7×10^{-5}		1.2×10^{-4}	6.1×10^{-9}
Other aluminum-clad targets (Table E-3)							
Population	3.7×10^{-6}	NA	NA	NA	NA	2.6×10^{-2}	3.7×10^{-6}
Maximally exposed individual	4.9×10^{-9}					3.6×10^{-6}	4.9×10^{-9}
Uninvolved worker	6.1×10^{-9}					1.2×10^{-4}	6.1×10^{-9}
TRR fuel and EBR-II slugs[b] (Table E-12)							
Population	(a)	8.8×10^{-2}	8.8×10^{-2}	NA	8.8×10^{-2}	8.8×10^{-2}	4.8×10^{-9}
Maximally exposed individual	6.5×10^{-9}	1.5×10^{-5}	1.5×10^{-5}		1.5×10^{-5}	1.5×10^{-5}	6.5×10^{-9}
Uninvolved worker	1.5×10^{-6}	2.6×10^{-4}	2.6×10^{-4}		2.6×10^{-4}	2.6×10^{-4}	1.5×10^{-6}

L11-3
TC

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a. These data are not available.

b. TRR = Taiwan Research Reactor; EBR = Experimental Breeder Reactor.

for the most affected sector (the northwest direction) of the 80-kilometer (50-mile) population. An examination of the distribution of communities of low-income persons and people of color did not indicate that there would be high and disproportionate impacts from potential actions.

In addition, Table 4-31 lists the point estimate of increased risk of latent cancer fatalities resulting from the calculated population dose for the accident that poses the greatest risk (i.e., the accident that has the highest product when the population dose is multiplied by the accident frequency). Because this projected point estimate of increased risk considers the projected accident probability, it provides a more appropriate index of the hazard associated with each material and scenario.

According to the Environmental Protection Agency, the average annual cancer fatality risk to an TC individual is approximately 0.0019. The annual cancer fatality risk estimates listed in Table 4-31 represent the additional (or incremental) risk from an accident involving the management of nuclear materials. For example, if the highest consequence accident involving plutonium-242 under the Continuing Storage Alternative were to occur, the estimated additional individual risk would be 0.000015; therefore, the "total" risk to that individual would be the national average risk of 0.0019 plus 0.000015, or 0.001915. This increase in risk would be small in comparison to the national average; furthermore, DOE could reduce the additional risk from accidents involving nuclear materials by stabilizing the materials. Such a reduction would occur because the likelihood of some accidents is small and because the amount of radioactive material released could decrease for some accidents. Solutions stored in locations not designed for long-term storage are examples of materials that would offer such benefits if solidified and packaged properly.

DOE evaluated the impacts associated with hazardous or toxic chemicals for each entire facility that would be involved in the storage or stabilization of nuclear materials rather than attempting to attribute the hazardous chemicals to the specific nuclear material process or activity the chemical supports. The approach used in this EIS for determining hazardous chemical impacts is similar to that typically used in a facility hazard assessment. Each facility was assumed to contain its maximum chemical inventory, which in turn was assumed to be totally released to the environment without TE postulating accident scenarios or release mechanisms. This approach provides results that bound all alternatives and scenarios. Appendix E presents the hazardous chemical impacts associated with this bounding condition.

As with radiological accidents, impacts to a close-in worker from a chemical accident can be severe or TE life-threatening. Some instances (i.e., the total releases of the hydrofluoric or nitric acid inventory) could exceed the chemical emergency response threshold values for uninvolved workers. These threshold values could be life-threatening if individuals were exposed for longer than 1 hour. However, because these individuals would be notified and evacuated within 1 hour of an inadvertent release, DOE does not expect any life-threatening or long-term effects. The projected maximum chemical concentration at the Site boundary could exceed the first emergency response level for nitric acid. The short-term health effects from this level of exposure would be irritation of the eyes and an objectionable odor. If DOE implemented the preferred alternative for each nuclear material, the need for chemicals to support storage or processing of these materials would diminish over the 10-year period covered by this EIS.

As stated in Chapter 1, one of the primary objectives of DOE's proposed action is to eliminate or reduce the risks from potential accidents that could be associated with the continued storage of nuclear materials at the SRS. For example, a wide range of accidents could result in the release of radioactive material from solutions currently stored in stainless-steel tanks that contain a variety of radioisotopes (plutonium-239, americium-243, curium-244, uranium-235, etc.).

4.2.2 RADIOLOGICAL IMPACTS FROM TRANSPORTATION ACCIDENTS

DOE used the RADTRAN computer program (Neuhauser and Kanipe 1992) to model radiological impacts from possible accidents involving onsite truck and rail movements of radioactive materials and wastes for each alternative. The program was configured with applicable SRS demographic data and transportation accident rates (HNUS 1994a,b). The parameters for the RADTRAN analysis included the transportation package dose rate, the number of packages per shipment, the number of shipments, the distance traveled, the fraction of travel in different population density zones, traffic counts, travel speed, and type of road traveled. The analyses did not include movements of radioactive material and waste within facility boundaries because they are part of each facility's accident analysis. TC

Radiological impacts (doses) were modeled for three affected human receptor groups: the onsite ("worker") population, members of the public residing near the SRS (the "offsite population"), and a hypothetical individual from the offsite population who would receive the greatest impact (the "maximally exposed individual," or MEI). The consequences associated with radioactive materials and radioactive waste shipments are expressed as excess latent cancer fatalities in each receptor group. Tables 4-32 and 4-33 list the analysis results for radioactive material shipments and radioactive waste shipments, respectively. DOE analyzed four categories of radioactive waste and has presented the

Table 4-32. Estimated latent cancer fatalities from medium probability accidents involving shipment of nuclear material.

Material	Accident probability[a]	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Vitrification (F-Canyon)	Processing and Storage for Vitrification (DWPF)	Improving Storage	
								Traditional construction schedule	Accelerated construction schedule
Stable materials		NT[b]	NA[c]	NA	NA	NA	NA	NA	NA
Plutonium-242	2.35×10^{-6}	NT	NT		NA		NT	NA	NA
Onsite population				0(d)		0			
Offsite population				0		0			
MEI[e]				0		0			
Americium and curium	2.35×10^{-6}		NA		NA		NT	NA	NA
Onsite population		0(f)		0		0			
Offsite population		0(f)		0		0			
MEI		0(f)		0		0			
Neptunium	3.33×10^{-5}	NT	NA		NA		NT	NA	NA
Onsite population				0		0			
Offsite population				0		0			
MEI				0		0			
H-Canyon plutonium-239 solutions	2.56×10^{-6}	NT			NA		NT	NA	NA
Onsite population			0	0		0			
Offsite population			0	0		0			
MEI			0	0		0			
H-Canyon enriched uranium solutions	2.56×10^{-5}	NT	NA	NT		NA	NT	NA	NA
Onsite population					1.11×10^{-3}				
Offsite population					8.21×10^{-5}				
MEI					1.08×10^{-8}				
Plutonium and uranium stored in vaults	2.05×10^{-5}								NA
Onsite population		0(g)	0	0		0	0	0	
Offsite population		0(g)	0	0		0	0	0	
MEI		0(g)	0	0		0	0	0	
Mark-31 targets	5.50×10^{-7}	NT					NT		
Onsite population			1.96×10^{-5}	1.96×10^{-5}		1.96×10^{-5}		1.96×10^{-5}	1.96×10^{-5}
Offsite population			1.12×10^{-6}	1.12×10^{-6}		1.12×10^{-6}		1.12×10^{-6}	1.12×10^{-6}
MEI			1.59×10^{-10}	1.59×10^{-10}		1.59×10^{-10}		1.59×10^{-10}	1.59×10^{-10}

TC

Table 4-32. (continued).

Material	Accident probability[a]	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Vitrification (F-Canyon)	Processing and Storage for Vitrification (DWPF)	Improving Storage: Traditional construction schedule	Improving Storage: Accelerated construction schedule
Mark-16 and -22 fuels	1.28×10^{-4}	NT	NA			NA			
Onsite population				1.11×10^{-3}	1.11×10^{-3}		1.11×10^{-3}	1.11×10^{-3}	1.96×10^{-5}
Offsite population				8.21×10^{-5}	8.21×10^{-5}		8.21×10^{-5}	8.21×10^{-5}	1.12×10^{-6}
MEI				1.08×10^{-8}	1.08×10^{-8}		1.08×10^{-8}	1.08×10^{-8}	1.59×10^{-10}
Other aluminum-clad targets	2.59×10^{-4}	NT	NA	NA	NA	NA			
Onsite population							1.96×10^{-5}	1.96×10^{-5}	1.96×10^{-5}
Offsite population							1.12×10^{-6}	1.12×10^{-6}	1.12×10^{-6}
MEI							1.59×10^{-10}	1.59×10^{-10}	1.59×10^{-10}
TRR/EBR-II[h] fuel	5.74×10^{-10}	NT			NA				
Onsite population			0	0		0	0	0	0
Offsite population			0	0		0	0	0	0
MEI			0	0		0	0	0	0

TC

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a. Cumulative probability.
b. NT = No transportation associated with the material for the given alternative.
c. NA = Not applicable.
d. A value of 0 is assigned for those alternatives in which the material is shipped in a container certified to contain its contents during an accident or the probability of an accident occurring is less than 1×10^{-7}.
e. MEI = Maximally exposed individual.
f. Under this alternative, the americium and curium targets and slugs in P-Reactor Basin are planned to be moved to the Receiving Basin for Offsite Fuel.
g. Under this alternative, the plutonium-238 solids in various site locations are planned to be moved to a vault in HB-Line.
h. TRR = Taiwan Research Reactor; EBR = Experimental Breeder Reactor.

Table 4-33. Estimated latent cancer fatalities from medium severity accidents involving shipment of radioactive waste.

Material	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Vitrification (F-Canyon)	Processing and Storage for Vitrification (DWPF)	Improving storage	
							Traditional construction schedule	Accelerated construction schedule
Stable materials		NA[a]	NA	NA	NA	NA	NA	NA
Onsite population	1.72×10^{-5}							
Offsite population	2.10×10^{-6}							
MEI[b]	3.00×10^{-10}							
Plutonium-242				NA			NA	NA
Onsite population	1.72×10^{-5}	18.4	18.4		18.4	1.72×10^{-5}		
Offsite population	2.10×10^{-6}	2.02	2.02		2.02	2.1×10^{-6}		
MEI	3.00×10^{-10}	2.89×10^{-4}	2.89×10^{-4}		2.89×10^{-4}	3.00×10^{-10}		
Americium and curium		NA		NA			NA	NA
Onsite population	1.72×10^{-5}		1.72×10^{-5}		1.72×10^{-5}	1.72×10^{-5}		
Offsite population	2.10×10^{-6}		2.10×10^{-6}		2.10×10^{-6}	2.10×10^{-6}		
MEI	3.00×10^{-10}		3.00×10^{-10}		3.00×10^{-10}	3.00×10^{-10}		
Neptunium		NA		NA			NA	NA
Onsite population	1.72×10^{-5}		18.4		1.72×10^{-5}	1.72×10^{-5}		
Offsite population	2.10×10^{-6}		2.02		2.10×10^{-6}	2.10×10^{-6}		
MEI	3.00×10^{-10}		2.89×10^{-4}		3.00×10^{-10}	3.00×10^{-10}		
H-Canyon plutonium-239				NA			NA	NA
solutions	1.72×10^{-5}	18.4	18.4		1.72×10^{-5}	1.72×10^{-5}		
Onsite population	2.10×10^{-6}	2.02	2.02		2.10×10^{-6}	2.10×10^{-6}		
Offsite population	3.00×10^{-10}	2.89×10^{-4}	2.89×10^{-4}		3.00×10^{-10}	3.00×10^{-10}		
MEI								
H-Canyon enriched uranium		NA			NA		NA	NA
solutions	1.72×10^{-5}		1.72×10^{-5}	1.72×10^{-5}		1.72×10^{-5}		
Onsite population	2.10×10^{-6}		2.10×10^{-6}	2.10×10^{-6}		2.10×10^{-6}		
Offsite population	3.00×10^{-10}		3.00×10^{-10}	3.00×10^{-10}		3.00×10^{-10}		
MEI								
Plutonium and uranium stored				NA				NA
in vaults	18.4	18.4	18.4		18.4	18.4	18.4	
Onsite population	2.02	2.02	2.02		2.02	2.02	2.02	
Offsite population	2.89×10^{-4}	2.89×10^{-4}	2.89×10^{-4}		2.89×10^{-4}	2.89×10^{-4}	2.89×10^{-4}	
MEI								

TC

Table 4-33. (continued).

Material	Continuing Storage (No Action)	Processing to Metal	Processing to Oxide	Blending Down to Low Enriched Uranium	Vitrification (F-Canyon)	Processing and Storage for Vitrification (DWPF)	Improving storage	
							Traditional construction schedule	Accelerated construction schedule
Mark-31 targets				NA				
Onsite population	1.72×10^{-5}	18.4	18.4		18.4	1.72×10^{-5}	1.72×10^{-5}	1.72×10^{-5}
Offsite population	2.10×10^{-6}	2.02	2.02		2.02	2.10×10^{-6}	2.10×10^{-6}	2.10×10^{-6}
MEI	3.00×10^{-10}	2.89×10^{-4}	2.89×10^{-4}		2.89×10^{-4}	3.00×10^{-10}	3.00×10^{-10}	3.00×10^{-10}
Mark-16 and -22 fuels		NA			NA			
Onsite population	1.72×10^{-5}		1.72×10^{-5}	1.72×10^{-5}		1.72×10^{-5}	1.72×10^{-5}	1.72×10^{-5}
Offsite population	2.10×10^{-6}		2.10×10^{-6}	2.10×10^{-6}		2.10×10^{-6}	2.10×10^{-6}	2.10×10^{-6}
MEI	3.00×10^{-10}		3.00×10^{-10}	3.00×10^{-10}		3.00×10^{-10}	3.00×10^{-10}	3.00×10^{-10}
Other aluminum-clad targets		NA	NA	NA	NA			
Onsite population	1.72×10^{-5}					1.72×10^{-5}	1.72×10^{-5}	1.72×10^{-5}
Offsite population	2.10×10^{-6}					2.10×10^{-6}	2.10×10^{-6}	2.10×10^{-6}
MEI	3.00×10^{-10}					3.00×10^{-10}	3.00×10^{-10}	3.00×10^{-10}
TRR/EBR-II[c] fuel				NA				
Onsite population	1.72×10^{-5}	18.4	18.4		18.4	1.72×10^{-5}	1.72×10^{-5}	1.72×10^{-5}
Offsite population	2.10×10^{-6}	2.02	2.02		2.02	2.10×10^{-6}	2.10×10^{-6}	2.10×10^{-6}
MEI	3.00×10^{-10}	2.89×10^{-4}	2.89×10^{-4}		2.89×10^{-4}	3.00×10^{-10}	3.00×10^{-10}	3.00×10^{-10}

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a. NA = Not applicable.

b. MEI = Maximally exposed individual.

c. TRR = Taiwan Research Reactor; EBR = Experimental Breeder Reactor.

waste type with the greatest consequence for each alternative in Table 4-33. For the transportation of both radioactive materials and radioactive waste, the data represent the maximum reasonably foreseeable accident.

DOE used the AXAIR89Q (Hamby 1994) computer program with SRS-specific meteorological data to model offsite consequences from releases of radioactivity from postulated onsite accidents. AXAIR89Q conservatively calculates maximally exposed individual and population doses because it uses air dispersion parameters that are exceeded less than 0.5 percent of the time.

The magnitude of accident consequences would depend on the amount of radioactive material released to the environment to which receptors would be exposed, the duration of the exposure, and the number of people exposed. DOE considered both the consequences and the probability of vehicle accidents in the transportation impacts model. The Department based its calculation of the joint probability of a given severity of accident occurring for each type of radioactive material and waste shipped on the probability of a range of impact forces that a package could receive in a hypothetical accident (NRC 1977), vehicle accident rates, and number of miles traveled.

TC The amount of radioactive material released to the environment, and hence the severity of an accident, is determined by the amount of damage to the package. The dispersion of radioactive material downwind from the damaged package and the fraction of airborne material that is of a size capable of being inhaled are modeled to calculate the amount of radioactive material to which the receptors are exposed.

For many of the radioactive materials listed in Table 4-30, the projected excess latent cancer fatalities would be negligible because the materials would be transported in containers designed for full containment. However, the DOE analysis showed potential health effects from some of the postulated accidents listed in Tables 4-32 and 4-33. These accidents could release some radioactive material because the transport package would not necessarily maintain full containment under the most severe accident. For the materials listed in Table 4-32, the calculated range of nonzero consequences for the on- and offsite populations could result in approximately 2×10^{-5} to 1×10^{-3} excess latent cancer fatality and approximately 1×10^{-6} to 8×10^{-5} excess latent cancer fatality, respectively. Similarly, for the radioactive waste listed in Table 4-33, the calculated range of nonzero consequences for the on- and offsite populations would result in approximately 2×10^{-5} to 2×10^{1} latent cancer fatality and approximately 2×10^{-6} to 2 latent cancer fatalities, respectively. The transportation of transuranic wastes could pose greater health risks to the on- and offsite populations than the transportation of other wastes. DOE recognizes, however, that these potentially greater risks would be the result of postulated accidents that have small chances of occurring (on the order of less than 1 in 10 million).

4.2.3 ENVIRONMENTAL JUSTICE ACCIDENT ASSESSMENT

The 99.5-percent meteorology model indicates that the SRS sector most affected by radiological accidents is the northwest. Although this is not typical of weather conditions (i.e., not the prevailing wind direction), the model calculated the highest impact to an individual at the SRS boundary. Figures 3-7 and 3-8 show the distributions, by census tracts, of people of color and low-income populations, respectively. Parts of two census tracts in the Northwest sector adjoin the SRS. Neither tract is a low-income community or a community comprised of 50 percent or more of people of color, although one of the tracts contains between 35 and 50 percent of color.

Figure 4-2 and Table 4-28 show the distribution of impacts from normal radiological releases that is more representative of typical weather conditions. If this same model were used to estimate accident doses, the values in the table would increase proportionally, because the only difference would be the source term (i.e., the amount and type of radioactive material). As a result of these proportional impacts, the conclusions drawn about disproportionate impacts are the same for normal or accidental radiological releases. Therefore, the accident scenarios would not result in either disproportionately high or adverse human health and environmental impacts on people of color or low-income populations. TC

4.3 Impacts Associated with Construction of a Dry Storage Facility

Chapter 3 and Appendix C contain descriptions of a proposed Dry Storage Facility that DOE could construct at the SRS to accommodate fuel and target elements for an extended period. Sections 4.1 and 4.2 describe the impacts of operating such a facility. This section describes the impacts of constructing the facility on a representative host site.

Appendix C describes two possible configurations for the Dry Storage Facility: a dry vault design and a dry cask design. The discussions in this chapter are relatively independent of the design DOE would choose because the two designs would produce only slight differences in impacts.

4.3.1 LAND USE

The Dry Storage Facility would be in an undeveloped area that at present is covered by timber. Construction activities, including laydown areas, would disturb about 0.08 square kilometer (20 acres) of land. After construction, the facility would occupy about 0.03 square kilometer (8 acres). Construction is not likely to affect land use patterns on the SRS.

4.3.2 SOCIOECONOMICS

Section 2.7 discusses the capital costs of constructing the Dry Storage Facility. The relative socioeconomic impact from construction expenditures and construction employment on the region of influence around the site would be negligible because SRS employees probably would perform the construction.

4.3.3 CULTURAL RESOURCES

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A 1990 Programmatic Memorandum of Agreement between the DOE Savannah River Field Office, the South Carolina Historic Preservation Officer (SHPO), and the Advisory Council on Historic Preservation is the instrument for the management of cultural resources at the SRS. DOE uses this agreement to identify cultural resources, assess them in terms of eligibility for the National Register of Historic Places, and develop mitigation plans for affected resources in consultation with the SHPO. DOE would comply with the stipulations of the memorandum of agreement for all activities related to the construction of the Dry Storage Facility.

There are no known cultural or historic resources on the host site. In addition, the area has a low archaeological site density. Therefore, activities in this zone would have a low probability of encountering archaeological sites and virtually no chance of affecting large sites with more than three prehistoric components. DOE has not initiated a survey of the host site for archaeological resources, but would do so before the start of any construction-related activities.

Three Native American groups have expressed concerns about the possible existence on the SRS of several plant species traditionally used in Tribal ceremonies. These plant species are known to occur on the Site, typically in wet sandy areas such as evergreen shrub bogs and savannas. These plants are not likely to occur in the host site due to the lack of suitable habitat.

4.3.4 AESTHETICS AND SCENIC RESOURCES

The construction of the Dry Storage Facility would not cause adverse impacts to aesthetic or scenic resources. The facility would not be visible from any onsite or offsite public access roads. Potential soil erosion and dust generation associated with construction-related activities would be controlled by the implementation of best management practices. Any visibility impacts from fugitive dust generation caused by construction-related activities would be minor and of short duration.

4.3.5 GEOLOGY

No unique features or minerals of economic value on the SRS would receive adverse impacts from site development. The construction of the Dry Storage Facility would result in localized impacts to surficial soils and would necessitate clearing and grading of 0.036 square kilometer (9 acres) of land. Site preparation, land shaping, and grading activities associated with construction would present a slight to moderate erosion hazard, which would be controlled and minimized by the implementation of best management practices.

4.3.6 AIR QUALITY

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There would be no radiological emissions from construction activities associated with the Dry Storage Facility. Potential air quality impacts from construction would include the generation of fugitive dust (particulate matter), smoke from earth moving and clearing operations, and emissions from the construction equipment. In addition, cleared vegetation would be burned at the construction site rather than hauled to a landfill. Sources of fugitive dust would include:

- Transfer of soil to and from haul trucks and storage piles
- Turbulence created by construction vehicles moving over cleared unpaved surfaces
- Wind-induced erosion of exposed surfaces

The amount of fugitive dust emitted during construction activities would be proportional to the amount of land cleared and the amount of vegetation burned. These emissions were calculated using EPA AP-42 emission factors (EPA 1985) for an assumed 0.04 square kilometer (10 acres) of land to be cleared. DOE calculated maximum downwind concentrations at the SRS boundary for total suspended particulates and particulate matter less than 10 microns using the EPA TSCREEN model (EPA 1988).

DOE estimated combustion emissions for the construction of the Dry Storage Facility. These emissions would originate from the operation of trucks, graders, roller/compactors, asphalt machines, etc. DOE used EPA factors for each type of equipment (EPA 1981) to derive the emission factors and estimated maximum boundary line concentrations using the TSCREEN air dispersion model (EPA 1988).

Table 4-34 compares maximum estimated downwind concentrations for pollutants emitted during construction activities and South Carolina's ambient air quality standards. As listed, air quality impacts associated with construction-related activities would not exceed the State standards. Therefore, construction activities would not have any detrimental effect on the health and safety of the general population.

Table 4-34. Air pollutant concentrations (micrograms per cubic meters) due to construction activities.

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Pollutant	Ambient standard	Concentration due to construction activities
Total suspended particulate - annual average	75	0.12
Particulate matter - 24-hour average	150	6.0
Particulate matter - annual average	50	0.08
Carbon monoxide - 1-hour average	40,000	190
Carbon monoxide - 8-hour average	10,000	130
Nitrogen oxides - annual average	100	0.035
Sulfur dioxide - 3 hour average	1,300	8.2
Sulfur dioxide - 24-hour average	365	1.1
Sulfur dioxide - annual average	80	0.003

4.3.7 WATER RESOURCES

Best management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the SRS. The estimated total water usage during the construction of the Dry Storage Facility would be about 50 million liters (13 million gallons). By comparison, the annual average water usage at SRS is almost 90 billion liters (23 billion gallons). Therefore, construction activities over 5 years would use much less water than the Site currently uses in 1 year. DOE anticipates minimal impacts on SRS water resources.

4.3.8 ECOLOGY

4.3.8.1 Terrestrial Resources

The host site for the Dry Storage Facility encompasses undeveloped forest land. Surface vegetation consists primarily of upland pine stands. Loblolly and slash pine dominate, but small pockets of hardwoods (oak, hickory, sweetgum, and yellow poplar) are evident. The location possesses suitable habitat for white-tailed deer and feral hogs as well as other faunal species common to the mixed pine/hardwood forests of South Carolina. The construction of the Dry Storage Facility would necessitate the clearing of 0.08 square kilometer (20 acres) and, therefore, would not be likely to affect the terrestrial ecology of the area.

4.3.8.2 Wetlands

There are two small wetland areas on the perimeter of the host site, but there is sufficient land area at the location to avoid these habitats. The implementation of best management practices to control surface runoff and sedimentation would ensure the protection of these wetlands and the aquatic ecosystem during construction activities.

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4.3.8.3 Threatened and Endangered Species

The host site contains no suitable habitat for threatened, endangered, or candidate species known to occur on or near the SRS. The southern bald eagle and wood stork feed and nest near wetlands, streams, and reservoirs, and thus would not be attracted to the construction site. Red-cockaded woodpeckers prefer open pine forests with mature trees more than 70 years old for nesting and 30 years old for foraging. This species does not use the relatively young pine stands (5 to 40 years) at the host site. The nearest red-cockaded woodpecker colony is across Upper Three Runs Creek, approximately 3.2 kilometers (2 miles) north of H-Area. Impacts to threatened and endangered species are not anticipated.

4.3.9 NOISE

Noise generated on the SRS by the construction of the Dry Storage Facility should not cause adverse impacts to the public or the environment because the noise would be site-specific and short-lived. There could be a slight increase in truck traffic to and from the host site, but that is not likely to result in a perceptible increase in traffic noise or any change in community reaction to noise.

4.3.10 TRAFFIC AND TRANSPORTATION

Construction materials, wastes, and excavated materials would be transported both on and off the SRS. These activities would result in increases in operation of personal-use vehicles by commuting construction workers, commercial truck traffic, and in traffic associated with daily operations of the SRS. The temporary increase in worker and materials traffic would be small in comparison to existing traffic loads. Traffic congestion would be minimal. Assuming the commercial trucks comply with Federal and State loading and speed regulations, damage to the roadbed would be minimal. TC

4.3.11 UTILITIES AND ENERGY

The total estimated utility and energy requirements during construction would be 8.5 megawatt-hours of electricity, 320,000 liters (85,000 gallons) of fuel, and 50 million liters (13 million gallons) of water. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Net sitewide increases in consumption would be minimal because overall activity on the SRS is likely to decrease due to changes in the Site mission.

CHAPTER 5. CUMULATIVE IMPACTS

Because DOE must select and implement an alternative for each material, this chapter has been organized into two sections. Section 5.1 describes the cumulative impacts that could result from a hypothetical combination of actions anaylzed in this EIS. Section 5.2 describes the cumulative impacts that could result from other actions proposed by DOE that potentially involve the SRS, and the impacts from commercial nuclear facilities adjacent to the SRS.

5.1 Cumulative Impacts of DOE Actions Associated with the Interim Management of Nuclear Materials

The number of materials and the number of reasonable alternatives for managing each material can lead to a large number (more than 5,000,000) of possible combinations that DOE could select. Because the presentation of so many combinations is impractical, DOE developed four hypothetical scenarios – No Action, Minimum Processing, Preferred, and Comparative – to illustrate the range of cumulative impacts that could result. Table 5-1 lists the alternatives that comprise these scenarios. Table 5-2 lists environmental impact data for the scenarios. Chapters 2 and 4 and Appendixes D and E present the relative impacts of all alternatives for a particular material. TC L11-3

In addition to the impacts listed in Table 5-2, Table 5-3 lists the impacts that would be related to the construction of a Dry Storage Facility based on the discussion in Section 4.3. The combination of particular alternatives to form a scenario considers the data in Table 5-3 if the scenario includes the Improving Storage Alternative for one of the fuel and target categories (Mark-31 targets, Mark-16 and -22 fuels, other aluminum-clad targets, Taiwan Research Reactor fuel, and Experimental Breeder Reactor-II slugs). As discussed in Chapter 4, the impacts associated with the modification of facilities or the construction of the Actinide Packaging and Storage Facility would be minor; therefore, Table 5-3 does not list them separately.

The four scenarios cover the spectrum of alternatives and illustrate the contrast between the smallest impacts and the largest impacts that could result. For each environmental factor, DOE summed the 10-year impacts from the alternatives corresponding to the given scenario.

- **No-Action Scenario** (see Section 2.5.3.1) - The impacts projected for this scenario could occur if current storage practices continued over the 10-year period. There is, however, a degree of uncertainty associated with these projections in terms of increased impacts for such factors as worker and population radiation exposure, which are dictated by the performance

Table 5-1. Composition of management scenarios.

Material	No Action	Minimum Processing	Preferred	Comparative
Stable material	Continuing Storage	Continuing Storage	Continuing Storage	Continuing Storage
Plutonium-242	Continuing Storage	Processing to Oxide	Processing to Oxide	Vitrification (F-Canyon)
Americium and curium	Continuing Storage	Solution - Vitrification (F-Canyon) Targets - Continuing Storage	Solution - Vitrification (F-Canyon) Targets - Continuing Storage	Processing to Oxide
Neptunium	Continuing Storage	Processing to Oxide	Processing to Oxide	Processing to Oxide
H-Canyon plutonium-239 solutions	Continuing Storage	Processing to Oxide	Processing to Oxide	Processing to Metal
H-Canyon enriched uranium solutions	Continuing Storage	Blending Down to Low Enriched Uranium	Blending Down to Low Enriched Uranium	Blending Down to Low Enriched Uranium
Plutonium and uranium stored in vaults	Continuing Storage	Improving Storage	Processing to Metal[a] Improving Storage[a] Processing to Oxide[a] Vitrification (F-Canyon)[a]	Vitrification (F-Canyon)
Mark-31 targets	Continuing Storage	Improving Storage	Processing to Metal	Vitrification (F-Canyon)
Mark-16 and -22 fuels	Continuing Storage	Improving Storage	Continuing Storage	Processing and Storage for Vitrification (DWPF)[b]
Other aluminum-clad fuel and targets	Continuing Storage	Improving Storage	Continuing Storage	Processing and Storage for Vitrification (DWPF)
Failed TRR fuel and EBR-II slugs[c]	Continuing Storage	Improving Storage	Processing to Metal	Vitrification (F-Canyon)

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a. For the plutonium and uranium stored in vaults, there are four preferred alternatives. DOE will choose the appropriate alternative for a particular solid based on results of the material inspection, as discussed in Section 2.3.3. The analysis in this EIS presents the impacts from Processing to Metal (which would produce the greatest impacts) as a conservative estimate of impacts.

b. DWPF = Defense Waste Processing Facility.

c. TRR = Taiwan Research Reactor; EBR = Experimental Breeder Reactor.

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characteristics of the stored material. For example, the continued degradation of fuel or targets in the SRS reactor basins would result in the release of more fission products to the basin water, which in turn could result in higher worker radiation exposures. Experience with the long-term storage of degrading fuel or other potentially unstable material such as plutonium or americium and curium solutions is limited and makes the prediction of future impacts difficult. The probability of accidents resulting from rapid chemical or physical changes in a material could increase over time (e.g., a criticality due to precipitation of fissile material in a tank). However, the consequences (exposures and health effects) can be

Table 5-2. Data for impact analysis of potential scenarios of alternatives.

Factor	Scenario			
	No Action	Minimum Processing	Preferred	Comparative
Atmospheric MEI[a] dose (rem) - 10-year total	8.4×10^{-5}	2.0×10^{-3}	5.4×10^{-3}	5.6×10^{-3}
Liquid MEI dose (rem) - 10-year total	2.0×10^{-4}	1.1×10^{-4}	1.6×10^{-4}	1.8×10^{-4}
Total MEI dose (rem) - 10-year total	2.9×10^{-4}	2.2×10^{-3}	5.6×10^{-3}	5.8×10^{-3}
Atmospheric population dose (person-rem) - 10-year total	3.8	84	220	230
Liquid population dose (person-rem) - 10-year total	0.75	0.49	0.65	0.71
Total population dose (person-rem) - 10-year total	4.6	84	220	230
Offsite CO[b] concentration ($\mu g/m^3$) - 1-hour average	9.6	33	47	58
Offsite CO concentration ($\mu g/m^3$) - 8-hour average	2.3	8.0	11	14
Offsite NO_x(c) concentration ($\mu g/m^3$) - annual average	0.19	0.62	1.7	2.2
Offsite SO_2(d) concentration ($\mu g/m^3$) - 3-hour average	5.6×10^{-3}	0.019	0.027	0.034
Offsite SO_2 concentration ($\mu g/m^3$) - 24-hour average	1.3×10^{-3}	4.4×10^{-3}	6.1×10^{-3}	7.6×10^{-3}
Offsite SO_2 concentration ($\mu g/m^3$) - annual average	7.9×10^{-5}	2.8×10^{-4}	3.8×10^{-4}	4.8×10^{-4}
Offsite gaseous fluorides ($\mu g/m^3$) - 12-hour average	0.016	0.070	0.28	0.36
Offsite gaseous fluorides ($\mu g/m^3$) - 24-hour average	8.6×10^{-3}	0.038	0.15	0.19
Offsite gaseous fluorides ($\mu g/m^3$) - 1-week average	3.4×10^{-3}	0.015	0.058	0.075
Offsite gaseous fluorides ($\mu g/m^3$) - 1-month average	9.5×10^{-4}	4.2×10^{-3}	0.016	0.021
Offsite HNO_3(e) concentration ($\mu g/m^3$) - 24-hour average	0.27	1.1	4.1	5.3
Offsite HNO_3 concentration ($\mu g/m^3$) - annual average	0.018	0.074	0.28	0.36
Onsite CO concentration (mg/m^3) - 8-hour average	0.015	0.053	0.074	0.092
Onsite NO_x concentration (mg/m^3) - 1-hour average	0.11	0.36	1.1	1.3
Onsite SO_2 concentration (mg/m^3) - 8-hour average	2.2×10^{-5}	7.7×10^{-5}	1.1×10^{-4}	1.3×10^{-4}
Onsite HNO_3 concentration (mg/m^3) - 8-hour average	4.2×10^{-3}	0.017	0.065	0.084
Onsite CO_2 concentration (mg/m^3) - 8-hour average	1.1×10^{-5}	3.8×10^{-5}	5.3×10^{-5}	6.6×10^{-5}
Average number of radiation workers	1,479	1,258	1,648	2,043
Collective worker dose (person-rem)	461	933	1,405	1,936
Water usage (millions of liters)	39,000	30,000	40,000	51,000
Electricity usage (megawatt-hour)	1,300,000	880,000	1,100,000	1,400,000
Steam usage (millions of kilograms)	6,000	3,700	4,600	5,900
Fuel usage (thousands of liters)	36,000	22,000	28,000	36,000
High-level liquid waste generation (millions of liters)	30	31	43	55
Equivalent DWPF[f] canisters	200	170	310	1,400
Saltstone generation (cubic meters)	35,000	38,000	70,000	100,000
TRU[g] waste generation (cubic meters)	830	1,300	1,800	1,800
Hazardous/mixed waste generation (cubic meters)	1,100	1,400	1,900	1,800
Low-level waste generation (cubic meters)	140,000	140,000	140,000	160,000

a. MEI = Maximally exposed offsite individual.
b. CO = Carbon monoxide.
c. NO_x = Nitrogen oxides.
d. SO_2 = Sulfur dioxide.
e. HNO_3 = Nitric acid.
f. DWPF = Defense Waste Processing Facility.
g. TRU = Transuranic.

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L11-3
L11-4
L11-22

Table 5-3. Environmental data associated with construction of the Dry Storage Facility.

Environmental Factor	Amount
Total suspended particulates - annual average (μg/m^3)	0.12
Particulate matter - 24-hour average (μg/m^3)	6.0
Particulate matter - annual average (μg/m^3)	0.08
Carbon monoxide - 1-hour average (μg/m^3)	190
Carbon monoxide - 8-hour average (μg/m^3)	130
Nitrogen oxides - annual average (μg/m^3)	0.035
Sulfur dioxide - 3 hour average (μg/m^3)	8.2
Sulfur dioxide - 24-hour average (μg/m^3)	1.1
Sulfur dioxide - annual average (μg/m^3)	0.003
Total water usage (liters[a])	50,000,000
Total fuel usage (liters)	320,000
Total electricity usage (megawatt-hours)	8.5

a. To convert liters to gallons, multiply by 0.26418.

predicted and DOE has used conservative assumptions in the associated analyses (see Appendix E).

TC
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L11-4

- **Minimum Processing Scenario** (see Section 2.5.3.3) - The impacts from this scenario would be the sum of the impacts from the minimal operation of the separations facilities that still achieved stabilization of all materials. For existing solutions of nuclear materials, radiochemical processing would be necessary to convert a solution to a stable form such as metal, oxide, or glass. DOE has chosen to include the appropriate preferred alternatives for solutions as part of this scenario. This scenario includes impacts from improving storage of other nuclear materials, including irradiated materials stored in basins and vault materials containing plutonium and uranium.

- **Preferred Scenario** (see Section 2.5.3.2) - The impacts from this scenario would be the sum of the impacts from the preferred alternatives for each type of material (as described in Chapter 2) over the 10-year period [i.e., No Action (Continuing Storage) for stable material + Plutonium-242 to Oxide + Americium/Curium Vitrification + Neptunium-237 to Oxide + H-Canyon Plutonium Solutions to Oxide + etc.]. These impacts are derived from data associated with similar or previous processing operations at the SRS.

- **Comparative Scenario** (see Section 2.5.3.4) - The estimated impacts from this scenario would be the highest overall for the 10-year period. DOE evaluated the alternatives for each type of material to determine those that would result in the highest overall impacts and then summed the impacts of the alternatives for presentation in the Comparative Scenario. DOE recognizes that the Comparative Scenario might not result in maximum impacts for every environmental factor considered; for example, an alternative for a given material could maximize worker and public health impacts but not those from radioactive waste generation. However, DOE believes that the Comparative Scenario is representative of the upper range of environmental impacts that could result from the selection of any other combination of alternatives.

5.1.1 PUBLIC AND WORKER HEALTH

Table 5-4 summarizes the estimated radiological health effects for each management scenario based on the data listed in Table 5-2. This summary data indicates that the estimated 10-year effects for the No-Action Scenario would have the smallest cumulative impact; those for both the Preferred Scenario and the Comparative Scenario, while higher than the No-Action Scenario, would be similar. The 10-year effects from the Minimum Processing Scenario would fall between the effects of the No-Action and the Preferred/Comparative Scenarios. The Minimum Processing Scenario would have 2 to 4 times less environmental impacts than either the Preferred or Comparative Scenarios, but about 4 to 10 times more environmental impacts than the No-Action Scenario.

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L11-4

Table 5-4. Estimated 10-year radiological health effects from normal operations.

	Scenario			
Subject	No Action	Minimum Processing	Preferred	Comparative
Public additional cancer deaths	0.0023	0.042	0.11	0.12
Worker additional cancer deaths	0.18	0.37	0.56	0.77
Probability of cancer death from MEI[a] dose	1 in 10 million	1 in 1 million	3 in 1 million	3 in 1 million
Probability of cancer death from worker maximum dose	3 in 1,000	3 in 1,000	3 in 1,000	3 in 1,000

a. MEI = Maximally exposed individual in the public.

The greatest calculated impact to the public could be 0.12 additional cancer death in the population within 80 kilometers (50 miles) of the Site, compared to a predicted 145,700 deaths from cancer due to all causes (23.5 percent of the population of 620,100; see Section 3.5.1) for the Comparative Scenario. The greatest calculated impact to workers could be 0.77 additional cancer death, compared to 480 cancers expected from all causes for the Comparative Scenario.

Under the No-Action Scenario, the lifetime impact on the public could be 0.0023 additional cancer death in the population within 80 kilometers (50 miles) of the Site. The lifetime impact to SRS workers involved with the No-Action Scenario could be 0.18 additional cancer death resulting from exposure to radiation over the 10-year period. The impacts on the maximally exposed individual and the maximally exposed worker are expressed not as latent cancer fatalities but as the additional lifetime probability of contracting a fatal cancer. For the maximally exposed member of the public, the additional or incremental probability of contracting a fatal cancer associated with the 10-year exposure to radiation would be 1 in 10 million. For the worker, the incremental probability would be 3 in 1,000.

The Preferred Scenario, the Comparative Scenario, and the Minimum Processing Scenario could increase the risk to the public. The estimated lifetime risk to the maximally exposed individual in the public from the 10-year exposure could increase to a maximum 3-in-1-million probability of contracting a fatal cancer. The estimated incremental risk for the maximally exposed worker would remain unchanged because administrative controls would limit maximum annual worker exposure.

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DOE evaluated the range of chemical concentrations to which the public and workers could be exposed due to nuclear material management activities, and expects minimal public and worker nonradiological health effects. DOE compared the data in Table 5-2 to Occupational Safety and Health Administration (OSHA) ceiling limits for protecting worker health and concluded that all concentrations are below these limits.

5.1.2 AIR RESOURCES

This section discusses radiological and nonradiological air quality impacts from normal operation for the four management scenarios described in Section 5.1.

Table 5-5 lists the doses from airborne releases of radioactivity associated with the management scenarios described in Section 5.1. For the No-Action Scenario, the estimated doses would remain constant over the 10-year period and within the 1993 totals from all SRS operations. The highest estimated annual dose to the maximally exposed member of the public associated with the No-Action Scenario, 0.0000084 rem (0.0084 millirem), would be less than 0.1 percent of the 10-millirem Environmental Protection Agency limit for sitewide airborne releases (40 CFR Part 61, Subpart H). The highest estimated annual population dose associated with the No-Action Scenario would be 0.38 person-rem.

Table 5-5. Estimated radiological doses from airborne releases of radioactivity associated with each management scenario.[a,b]

Receptor[c]	No Action	Minimum Processing	Preferred	Comparative
MEI[d] (rem)				
Maximum annual[e]	0.0000084	0.0018	0.0027	0.0035
10-year total	0.000084	0.0020	0.0054	0.0056
Population[f] (person-rem)				
Maximum annual[e]	0.38	75	110	144
10-year total	3.8	84	219	228

---

a. Based on data in Chapters 2 and 4.
b. Composite of all materials processed under that scenario.
c. Atmospheric releases from total 1993 SRS operations produced a dose of 0.00011 rem to the maximally exposed member of the public and 7.6 person-rem to the regional population (Arnett, Karapatakis, and Mamatey 1994).
d. Maximally exposed offsite individual.
e. The analysis first determined the maximum annual dose for each material among the treatment phases, and then summed the maximum doses for all materials to obtain an upper bound dose value.
f. Population within 80 kilometers (50 miles) of the SRS (regional population).

For the Minimum Processing Scenario, the highest annual incremental dose to the maximally exposed individual from airborne releases during the 10-year period could be 0.0018 rem (1.8 millirem). This incremental individual dose would represent 18 percent of the 10-millirem limit. The highest estimated annual incremental dose to the regional population from airborne releases during the 10-year period would be 75 person-rem.

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L11-4

For the Preferred Scenario, the highest annual incremental dose to the maximally exposed individual from airborne releases during the 10-year period could be 0.0027 rem (2.7 millirem). This incremental individual dose would represent 27 percent of the 10-millirem sitewide limit. The highest annual incremental dose to the regional population from airborne releases could be 109 person-rem.

For the Comparative Scenario, the highest estimated annual incremental dose to the maximally exposed individual from airborne releases during the 10-year period could be 0.0035 rem (3.5 millirem). This incremental individual dose would represent 35 percent of the 10-millirem limit. The highest estimated annual incremental dose to the regional population from airborne releases during this 10-year period would be 144 person-rem.

For nonradiological emissions, DOE compared the values in Table 5-2 to the nonradiological air quality standards listed in Table 5-10. As the data indicated, the potential concentrations would be below applicable standards.

5.1.3 WATER RESOURCES

This section describes the effects of liquid effluents from normal operations associated with the four management scenarios. Table 5-6 lists the maximum annual and total doses received from exposure to radionuclides in the Savannah River. For the No-Action Scenario, the estimated annual doses would remain constant over time. For the other management scenarios, estimated doses would generally increase during active phases of the stabilization activity (as shown in Appendix D) but would be less than the annual rate for No Action at the completion of stabilization. In other words, the estimated doses would decrease over time as the stabilization occurred.

Table 5-6. Estimated radiological doses from surface-water pathway exposures.

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	Scenario			
Dose[a]	No Action	Minimum Processing	Preferred	Comparative
MEI[b] (rem)				
Maximum annual[c]	0.000020	0.000039	0.000051	0.000088
10-year total	0.00020	0.00011	0.00016	0.00018
Population (person-rem)				
Maximum annual	0.075	0.15	0.19	0.37
10-year total	0.75	0.49	0.65	0.71

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a. Resulting from the use of Savannah River water between the SRS and the Atlantic Ocean.
b. MEI = Maximally exposed offsite individual.
c. The analysis first determined the maximum annual dose for each material among the treatment phases, and then summed the maximum doses for all materials to obtain an upper bound dose value.

5.1.4 WASTE GENERATION

Table 5-7 lists estimated volumes of high-level radioactive waste, low-level waste, saltstone, transuranic waste, and hazardous and mixed wastes projected to be generated by the scenarios described in Section 5.1. In addition, this table contains historic data and projected data to enable comparisons. The projected values assume the adoption of the Preferred Scenario.

Table 5-7. Estimated average annual waste generated over the 10-year time period by scenario.[a]

	Scenario					
Waste type	No Action	Minimum Processing	Preferred	Comparative	Historic values[b]	Projected values
High-level liquid waste (millions of liters per year)[c]	3.0	3.1	4.3	5.5	2.2	4.6[d]
Equivalent DWPF[e] canisters per year	20	17	31	140	(f)	405[g]
Saltstone (cubic meters per year)[h]	3,500	3,500	7,000	10,000	(f)	63,500[g]
Transuranic waste (cubic meters per year)	83	130	180	180	595	830[c]
Hazardous/mixed waste (cubic meters per year)	110	140	190	180	127	26,000[b,i]
Low-level waste (cubic meters per year)	14,000	14,000	14,000	16,000	18,500	18,000[b]

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a. Source: Based on data from WSRC (1995a).
b. Value(s) obtained from Waste Management Environmental Impact Statement (DOE 1995c).
c. To convert liters to gallons, multiply by 0.26418.
d. Most high-level liquid waste is associated with the storage, handling, and treatment of the materials discussed in this EIS.
e. DWPF = Defense Waste Processing Facility. TC
f. DWPF has not yet produced an actual canister of waste. L11-3
g. Yearly production rates from high-level waste tank inventory. This value is independent of the yearly generation rate of high-level waste. Approximately 130 million liters of liquid high-level waste are already in the F- and H-Area tanks (WSRC 1994g). L11-4
h. To convert cubic meters to yards, multiply by 1.3079.
i. The large increase in this value is dominated by potential decontamination and decommissioning efforts and environmental restoration activities that are scheduled to occur. This value also includes hazardous/mixed waste projections for the Defense Waste Processing Facility which is scheduled to begin operations during the 10-year period.

DOE estimated the sitewide 30-year expected waste generation forecast (DOE 1995c) from site operations, decontamination and decommissioning activities, and environmental restoration efforts. The total waste volume projected is 1.31×10^9 liters (1,310 million liters) or 1.23×10^8 cubic feet.

5.1.5 UTILITIES AND ENERGY

Table 5-8 compares the maximum annual 10-year cumulative consumption of utilities for all the scenarios to the current SRS utility capacities to determine the potential for impacts. Existing SRS capacities and distribution systems would be adequate to support any of the alternatives; no new generation or treatment facilities would be necessary. Suitable groundwater from the deep aquifers at the Site is abundant and aquifer depletion is not a problem. Pumping from the deep aquifer to meet

Table 5-8. Current and estimated utilities and energy usage associated for management scenarios.

TC L11-3 L11-4

Utility	Current SRS usage	Maximum from all scenarios
Electricity[a]	659,000 megawatt-hours per year	140,000 megawatt-hours per year
Water[b]	14.0 billion liters (3.0 billion gallons) of groundwater per year	5,100 million liters (1,350 million gallons) per year[c]
	75.7 billion liters (20.0 billion gallons) of surface water per year	
Fuel[a]	33.1 million liters (8.7 million gallons) per year[d]	3.6 million liters (950,000 gallons) per year

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a. Source: WSRC (1995a).
b. Source: DOE (1995f).
c. Includes both surface and groundwater usage.
d. Includes both fuel oil and gasoline.

TC domestic, process, and other water uses has continued as needed since the early 1950s. This usage has not adversely affected water levels in the deep aquifer (Christensen and Gordon 1983).

5.2 Cumulative Impacts of Other Reasonably Foreseeable Actions

TE This section presents the cumulative impacts of offsite (non-DOE) nuclear facilities and potential impacts of other contemplated DOE actions that would involve the Savannah River Site. Radiological impacts from the operation of the Vogtle Electric Generating Plant, a two-unit commercial nuclear powerplant approximately 16 kilometers (10 miles) southwest of the center of the SRS near Waynesboro, Georgia, are minimal, but DOE has factored them into the analysis. Radiological impacts from the operation of the Chem-Nuclear Services facility, a commercial low-level waste disposal facility just east of the SRS, are so small that they are not included in this assessment (SCDHEC 1992a).

In addition to the interim management of nuclear materials, DOE has recently prepared other National Environmental Policy Act (NEPA) documentation related to the Savannah River Site (see Section 1.6):

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- *Appendix C of the Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement* (DOE 1995f); the preferred alternative involves the shipment of aluminum-clad spent nuclear fuel to the SRS for storage and disposal.

- *Supplemental Environmental Impact Statement,* the SRS Defense Waste Processing Facility (DOE 1994g); the preferred alternative is the completion of construction and the operation of the DWPF to verify high-level radioactive waste at the SRS.

- *The F-Canyon Plutonium Solutions Environmental Impact Statement* (DOE 1994d); the preferred alternative involves the processing of F-Canyon plutonium solutions to metal.

- The *SRS Waste Management Final Environmental Impact Statement* (DOE 1995c); the preferred alternative involves the treatment and minimization of radioactive and hazardous wastes at the SRS.

- The *Draft Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel* (DOE 1995e); the proposed action could involve the shipment of foreign research reactor spent nuclear fuel to the SRS.

- The Draft Programmatic Environmental Impact Statement for Tritium Supply and Recycling (DOE 1995g); the proposed action could involve the construction of a reactor or accelerator for tritium production at the SRS, along with associated support facilities.

TC

DOE has decided to implement the preferred alternatives for all of the EISs listed above (see Section 1.6). The cumulative impact analysis in this chapter includes data associated with the preferred alternatives that DOE has now selected for the Final EISs listed. The potentially cumulative impacts of managing various types of spent fuel at the SRS were analyzed in the Programmatic Spent Nuclear Fuel Management EIS (DOE 1995f). DOE used data from the selected alternative, Regionalized Management by Spent Fuel Type (e.g., aluminum-clad fuels managed at the SRS), for cumulative impact analysis. This includes the potential management of spent fuel from foreign research reactors, which DOE is evaluating in the Foreign Research Reactor Spent Nuclear Fuel EIS.

The Programmatic EIS for Tritium Supply and Recycling is still in draft form and DOE has not identified a preferred alternative. Therefore, DOE used data associated with the alternative potentially the greatest environmental impacts discussed in the Programmatic EIS to estimate the cumulative impacts. Similarly, DOE used preliminary data associated with the alternative potentially resulting in the greatest environmental impacts from the Disposition of Surplus Highly Enriched Uranium EIS.

5.2.1 PUBLIC AND WORKER HEALTH

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Table 5-9 summarizes the cumulative health effects of routine SRS operations. The impacts from current SRS projects are based on 1993 data. Impacts resulting from proposed DOE actions are described in the environmental impact statements above. In addition to estimated radiological doses to the hypothetical maximally exposed individual and the offsite population, Table 5-9 lists potential cancer fatalities for the public and workers due to exposure to radiation.

5.2.2 AIR RESOURCES

TE

Table 5-10 compares the cumulative concentrations of nonradiological air pollutants from the SRS to Federal and state regulatory standards. The listed values are the maximum modeled concentrations that could occur at ground level at the Site boundary. The data demonstrate that total estimated concentrations of nonradiological air pollutants from the SRS would be well below the regulatory standards at the Site boundary.

TC

DOE also evaluated the cumulative impacts of airborne radioactive releases in terms of dose to a maximally exposed individual at the SRS boundary. DOE has included the impacts of the two-unit Plant Vogtle (NRC 1994) in this cumulative total. The radiological emissions from the operation of the Chem-Nuclear low-level waste disposal facility just east of the SRS are very low (SCDHEC 1992a), and are not included. Table 5-11 lists the results of this analysis, using 1993 emissions (1991 for Plant Vogtle) as the SRS baseline. The highest cumulative dose to the maximally exposed member of the public would be 0.0019 rem (or 1.9 millirem) per year, well below the regulatory standard of 10 millirem per year (40 CFR Part 61). Summing the doses to maximally exposed individuals for the five actions or facilities listed in Table 5-11 is an extremely conservative approach because it assumes that the maximally exposed individuals would occupy the same location over the same time period, which is a physical impossibility.

TC

Adding the population doses from current and projected activities at the SRS, including stabilization of plutonium solutions, operation of the proposed Defense Waste Processing Facility, tritium supply and recycle, and management of spent nuclear fuel could yield a total annual cumulative dose of 138 person-rem from airborne sources. This translates into 0.069 latent cancer fatality for each year of exposure of living population within an 80-kilometer (50-mile) radius of the SRS. For comparison, 145,700 deaths from cancer due to all causes would be likely in the same population over their lifetimes.

Table 5-9. Estimated average annual cumulative radiological doses and resulting health effects to offsite population and facility workers. TE

Activity	Maximally exposed individual				Total collective (to offsite population)[a]				Workers	
	Dose from airborne releases[b]	Dose from liquid releases[b]	Total dose[b]	Fatal cancer risk[c]	Dose from airborne releases[d]	Dose from liquid releases[d]	Total dose[d]	Latent cancer fatalities[e]	Dose[d]	Latent cancer fatalities[e]
Stabilization of plutonium solutions[f]	0.0000086	0.00000029	0.0000089	4.5×10^{-9}	0.38	0.0037	0.38	0.00019	131	0.052
Waste Management[g]	0.000031	6.9×10^{-7}	0.000031	1.6×10^{-8}	1.5	0.0068	1.5	0.00075	81	0.032
Defense Waste Processing Facility[h]	0.0000010	NA[i]	0.0000010	5.0×10^{-10}	0.07	NA	0.07	0.000035	118	0.047
Plant Vogtle[j]	0.00000037	0.00017	0.00017	8.5×10^{-8}	0.047	0.0097	0.057	0.000029	NA	NA
Surplus HEU disposition[k]	0.0000025	0	0.0000025	1.25×10^{-9}	0.16	0	0.16	0.00008	11	0.004
Tritium supply and recycling[l]	0.0015	0.000087	0.0016	7.9×10^{-7}	120	1.0	121	0.060	172	0.069
Spent nuclear fuel[m]	0.0004	0.0001	0.0005	2.5×10^{-7}	16.0	2.4	18.4	0.0092	79	0.032
Total	0.0019	0.00036	0.0023	0.0000012	138	3.4	142	0.071	592	0.24
1993 SRS practices[n]	0.00011	0.00014	0.00025	1.3×10^{-7}	7.6	1.5	9.1	0.0046	263	0.11

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a. Collective dose to the 80-kilometer population for atmospheric releases and to the downstream users of the Savannah River for liquid releases.
b. Dose in rem.
c. Probability of fatal cancer.
d. Dose in person-rem.
e. Incidence of excess fatal cancers.
f. Source: DOE (1994d).
g. Source: DOE (1995c).
h. Source: DOE (1994g).
i. NA = not applicable.
j. Source: NRC (1994).
k. HEU = highly enriched uranium; based on preliminary data.
l. Source: DOE (1995g).
m. Source: DOE (1995f); includes contribution from Foreign Research Reactor spent nuclear fuel.
n. Source: Arnett, Karapatakis, and Mamatey (1994); includes HB-Line operations.

TC

Table 5-10. Estimated maximum nonradiological cumulative ground-level concentrations of criteria and toxic pollutants (micrograms per cubic meter) at the SRS boundary.[a,b]

Pollutant	Averaging time	Regulatory standard	Cumulative concentration[c]
Carbon monoxide	1 hour	40,000	268 (0.67%)
	8 hours	10,000	38 (0.38%)
Nitrogen oxides	Annual	100	16 (16%)
Sulfur dioxide	3 hours	1,300	835 (64%)
	24 hours	365	199 (55%)
	Annual	80	15 (19%)
Gaseous fluorides	12 hours	3.7	1.1 (29%)
TC	24 hours	2.9	0.43 (15%)
	1 week	1.6	0.50 (31%)
	1 month	0.8	0.05 (6%)
Nitric acid	24 hours	125	5.7 (5%)

a. Sources: Hunter (1995a); DOE (1995f); DOE (1995g).
b. Numbers in parentheses indicate the percentage of the regulatory standard.
c. All SRS sources including Defense Waste Processing Facility operations, Consolidated Incineration Facility operations, spent nuclear fuel management (including foreign research reactor nuclear fuel), the stabilization of plutonium solutions in F-Canyon, the SRS waste management activities, HB-Line operations, tritium supply and recycling, and disposition of surplus highly enriched uranium (preliminary data).

Table 5-11. Estimated average annual cumulative radiological doses and resulting health effects to offsite population from airborne releases.

	Offsite population			
	Maximally exposed individual		Total collective (to 80-kilometer population)	
Activity	Dose[a]	Fatal cancer risk[b]	Dose[c]	Latent cancer fatalities[d]
Stabilization of F-Canyon plutonium solutions[e]	8.6×10^{-6}	4.3×10^{-9}	0.38	1.9×10^{-4}
TC Waste Management[f]	3.1×10^{-5}	1.6×10^{-8}	1.5	7.5×10^{-4}
Defense Waste Processing Facility[g]	1.0×10^{-6}	5.0×10^{-10}	0.07	3.5×10^{-5}
Plant Vogtle[h]	3.7×10^{-7}	1.9×10^{-10}	0.47	2.4×10^{-5}
Surplus HEU disposition[i]	2.5×10^{-6}	1.25×10^{-9}	0.16	8.0×10^{-5}
Tritium supply and recycling[j]	1.5×10^{-3}	7.5×10^{-7}	120	0.060
Spent nuclear fuel[k]	4.0×10^{-4}	2.0×10^{-7}	0.16	8.0×10^{-3}
TC Total	1.9×10^{-3}	9.7×10^{-7}	138	6.9×10^{-2}
1993 SRS practices[l]	1.1×10^{-4}	5.5×10^{-8}	7.6	3.8×10^{-3}

a. Dose in rem.
b. Probability of fatal cancer.
c. Dose in person-rem.
d. Incidence of excess fatal cancers.
e. Source: DOE (1995d); based on maximum annual releases.
f. Source: DOE (1995c)
g. Source: DOE (1994g).
h. Source: NRC (1994).
i. HEU = highly enriched uranium; based on preliminary data.
j. Source: DOE (1995g).
k. Source: DOE (1995f); includes contribution from Foreign Research Reactor Nuclear Fuel.
l. Source: Arnett, Karapatakis, and Mamatey (1994); includes HB-Line operations.

Environmental restoration, decontamination and decommissioning, tritium supply and recycling, and waste management activities and facilities that DOE is assessing in the SRS Waste Management EIS (DOE 1995c) would add variable increments to airborne emissions of radioactive and nonradioactive materials. TC

5.2.3 WATER RESOURCES

Table 5-12 summarizes the estimated cumulative radiological doses to human receptors from exposure to waterborne sources downstream from the Savannah River Site. Liquid effluents from the Site could contain small quantities of radionuclides that would be released to SRS streams that are tributaries of the Savannah River. The exposure pathways considered in this analysis included drinking water, fish ingestion, shoreline exposure, swimming, and boating. The ingestion of fish containing cesium-137 would contribute most of the exposure to both the maximally exposed individual and the offsite population. Plutonium and uranium isotopes ingested with drinking water would be secondary contributors. TC

Table 5-12. Estimated average annual cumulative radiological doses and resulting health effects to offsite population from liquid releases. TC

	Offsite population			
	Maximally exposed individual		Total collective (to downstream users of the Savannah River)	
Activity	Dose[a]	Fatal cancer risk[b]	Dose[c]	Latent cancer fatalities[d]
Stabilization of F-Canyon plutonium solutions[e]	2.9×10^{-7}	1.5×10^{-10}	0.0037	1.9×10^{-6}
Waste Management[f]	6.9×10^{-7}	3.5×10^{-10}	0.0068	3.4×10^{-6}
Plant Vogtle[g]	1.7×10^{-4}	8.5×10^{-8}	0.0097	4.9×10^{-6}
Surplus HEU disposition[h]	0	0	0	0
Tritium supply and recycling[i]	8.7×10^{-5}	4.4×10^{-8}	1.0	5.0×10^{-4}
Spent nuclear fuel[j]	1.0×10^{-4}	5.0×10^{-8}	2.4	1.2×10^{-3}
Total	3.6×10^{-4}	1.8×10^{-7}	3.4	1.7×10^{-3}
1993 SRS practices[k]	1.4×10^{-4}	7.0×10^{-8}	1.5	7.5×10^{-4}

a. Dose in rem.
b. Probability of fatal cancer.
c. Dose in person-rem.
d. Incidence of excess fatal cancers.
e. Source: DOE (1994d).
f. Source: DOE (1995c)
g. Source: NRC (1994).
h. HEU = highly enriched uranium; based on preliminary data.
i. Source: DOE (1995g).
j. Source: DOE (1995f); includes contribution from foreign research reactor nuclear fuel.
k. Source: Arnett, Karapatakis, and Mamatey (1994); includes HB-Line operations.

TC

The highest estimated cumulative dose to the maximally exposed member of the public from liquid releases would be 0.00036 rem (or 0.36 millirem) per year, well below the regulatory standard of 4 millirem per year (40 CFR Part 141). Adding the population doses from current and projected activities at the SRS, including the stabilization of plutonium solutions, operation of the proposed Defense Waste Processing Facility, and management of spent nuclear fuel, would yield a total estimated annual cumulative dose of 2.4 person-rem from liquid sources. This translates into 0.0013 latent cancer fatality for each year of exposure of the population living within an 80-kilometer (50-mile) radius of the SRS. For comparison, 15,300 deaths from cancer due to all causes would be likely in the population of 65,000 downstream residents over their lifetimes.

5.2.4 WASTE GENERATION

TC

Table 5-13 lists cumulative volumes of high-level radioactive waste, low-level waste, saltstone, transuranic waste, and hazardous and mixed wastes that the SRS would generate. The values are based on the SRS 30-year waste forecast (WSRC 1994g), the SRS Waste Management EIS (DOE 1995c), Appendix C to the Programmatic Spent Nuclear Fuel Management EIS (DOE 1995f), the Defense Waste Processing Facility Supplemental EIS (DOE 1994g), the F-Canyon Plutonium Solutions EIS (DOE 1994d), the Draft Programmatic EIS for Tritium Supply and Recycling (DOE 1995g), and preliminary data on the disposition of surplus highly enriched uranium.

Table 5-13. Estimated cumulative waste generation from SRS operations (cubic meters).[a,b]

Waste type	Volume[c]
High-level	2,000
Low-level	19,000
Saltstone	53,000
Transuranic	720
Mixed/hazardous	2,300

a. Average annual values based on waste forecast from 1995 to 2004.

b. To convert cubic meters to cubic yards, multiply by 1.3079.

TC

c. Includes proposed Defense Waste Processing Facility, Spent Nuclear Fuel management (low-level waste, high-level waste, and transuranic waste only), Stabilization of Plutonium Solutions in F-Canyon, HB-Line operations, tritium supply and recycling, and disposition of surplus highly enriched uranium (preliminary data).

5.2.5 UTILITIES AND ENERGY

Table 5-14 lists the cumulative consumption of electricity and water (surface water and groundwater) due to activities at the SRS.

Table 5-14. Estimated average annual cumulative utility consumption.

Activity	Electricity consumption (megawatt-hours)	Water usage[a] (liters)
Stabilization of F-Canyon plutonium solutions[b]	22,000	1.19×10^{9}
Defense Waste Processing Facility[c]	32,000	9.12×10^{7}
Surplus HEU disposition[d]	4,000	9.5×10^{7}
Tritium supply and recycling[e]	700,00	5.9×10^{10}
Spent nuclear fuel[f]	110,400	3.79×10^{8}
Total	170,000	1.76×10^{9}
1993 SRS usage[f]	659,000	8.97×10^{10}

TC

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a. Includes both groundwater and surface-water usage.
b. Source: DOE (1994d).
c. Source: DOE (1994g).
d. HEU = highly enriched uranium; based on preliminary data.
e. Source: DOE (1995g).
f. Source: DOE (1995f); includes contribution from Foreign Research Reactor Nuclear Fuel and HB-Line operations.

CHAPTER 6. SHORT-TERM USE VERSUS LONG-TERM RESOURCE COMMITMENTS

This section addresses the relationship between short-term uses of the environment and the maintenance of its long-term productivity.

Reinstituting activities at the F- and H-Canyons and support facilities to accommodate the management of nuclear materials would result in the short-term resource uses described in Chapter 4. However, these activities would not be likely to compromise environmental resources beyond the duration of management activities. As a result of normal operations, short-term use of the atmosphere as a receptor for emissions would have an incremental minimal effect on long-term global atmospheric conditions. DOE anticipates no increase in long-term resource commitments (e.g., electricity consumption).

The stabilization of nuclear materials involves tradeoffs between short- and long-term impacts. The selection of No Action for each material group discussed in this EIS probably would result in the smallest impacts over the 10-year period of analysis. The actions required to stabilize materials would entail some increased exposure and risk in comparison to No Action during the 10-year period. However, over the long term, No Action probably would produce greater impacts than those from the stabilization alternatives because it would result in the need for greater management vigilance and because there would be an increased probability that continued changes in material chemistry or degradation of the functions and physical structure of the facilities containing the materials could result in releases to the environment and consequent worker exposures. Furthermore, DOE eventually would have to take some kind of stabilization action, and the risks and exposures from such actions would occur at that time. TC

Processing alternatives would result in short-term exposures and risks compared to No Action. However, after the completion of stabilization activities, exposures and risk would be reduced in comparison to No Action, and materials could be consolidated in fewer facilities, further reducing exposures and risk. In addition, processing alternatives would generate waste, which would require management. Due to the current backlog of high-level waste at the SRS, the vitrification of such waste from the processing alternatives would not be completed for many years. As indicated in Chapter 4, DOE has estimated the number of Defense Waste Processing Facility canisters that would result from the implementation of processing alternatives and that would require placement in a geologic repository.

CHAPTER 7. IRREVERSIBLE OR IRRETRIEVABLE RESOURCE COMMITMENTS

Irreversible and irretrievable commitments of resources that could occur with the implementation of any of the alternatives to manage nuclear materials currently stored at the Savannah River Site fall into the categories of materials and energy. To a large extent the physical plant and facilities required to implement the alternatives being considered already exist in the F- and H-Areas, so the resources typically required to construct new buildings and establish new engineering processes would be relatively small. The exceptions to this would be modifications to produce the F-Canyon Vitrification Facility and the construction of a new Dry Storage Facility and an Actinide Packaging Facility. If necessary, the construction of such new facilities outside existing industrialized areas would require less than an estimated 0.4 square kilometer (100 acres) of onsite land.

7.1 Materials

The construction of the Actinide Processing Facility would require about 4,620 cubic meters (6,040 cubic yards) of concrete and about 1,775 metric tons (1,960 tons) of steel. The construction of the Dry Storage Facility would require about 17,950 cubic meters (23,520 cubic yards) of concrete and about 6,910 metric tons (7,600 tons) of steel.

Chemicals such as nitric acid and tributyl phosphate would be committed for the various alternative processes. The required chemicals and materials are readily available. Strategic and critical materials (e.g., beryllium, cadmium, cobalt) would not be required in quantities that would seriously reduce the national or world supply.

Existing facilities that DOE would use for management activities would have contaminated areas and equipment that would be unusable for recycling. This would include such materials as masonry, piping, metal structures and objects, flooring, and plastics.

7.2 Energy

All the alternatives would require power to operate the F- and H-Area buildings and the various process activities conducted in them. Steam would be used for applications such as evaporators and off-gas reactors. The fuel used to create electricity for the facilities would be purchased from commercial utilities. Small amounts of diesel fuel would also be used.

CHAPTER 8. APPLICABLE LAWS, REGULATIONS, AND OTHER REQUIREMENTS

8.1 Laws and Other Requirements

This section identifies and summarizes the major laws, regulations, Executive Orders, and DOE Orders that might apply to the interim management of nuclear materials.

Section 8.1.1 discusses the major Federal statutes that impose environmental protection and compliance requirements on DOE. In addition, there might be other State and local measures applicable to the interim management of nuclear materials because Federal law delegates enforcement or implementation authority to State or local agencies. Section 8.1.2 addresses environmentally related Executive Orders that clarify issues of national policy and set guidelines under which Federal agencies, including DOE, must act. DOE implements its responsibilities for protection of public health, safety, and the environment through a series of Departmental Orders that are mandatory for operating contractors of DOE-owned facilities. Section 8.1.3 discusses the DOE orders related to environmental, health, and safety protection.

TC

8.1.1 FEDERAL ENVIRONMENTAL STATUTES AND REGULATIONS

National Environmental Policy Act (NEPA) of 1969, as amended (42 USC 4321 et seq.)

NEPA establishes a national policy promoting awareness of the environmental consequences of the activity of humans on the environment and consideration of environmental impacts during the planning and decisionmaking stages of a project. This Act requires Federal agencies to prepare a detailed statement on the environmental effects of proposed major Federal actions that might significantly affect the quality of the human environment.

This EIS has been prepared in response to NEPA requirements and policies, and in accordance with Council on Environmental Quality (40 CFR Parts 1500 through 1508) and DOE regulations for implementing the procedural provisions of NEPA (10 CFR Part 1021). It discusses reasonable alternatives and their potential environmental consequences.

Atomic Energy Act of 1954, as amended (42 USC 2011 et seq.)

The Atomic Energy Act of 1954 authorizes DOE to establish standards to protect health or minimize dangers to life or property with respect to activities under its jurisdiction. Through a series of DOE Orders, DOE has established an extensive system of standards and requirements to ensure safe operation of its facilities.

Clean Air Act, as amended (42 USC 7401 et seq.)

The Clean Air Act, as amended, is intended to "protect and enhance the quality of the Nation's air resources so as to promote the public health and welfare and the productive capacity of its population." Section 118 of the Clean Air Act, as amended, requires each Federal agency, such as DOE, with jurisdiction over any property or facility that might result in the discharge of air pollutants, to comply with "all Federal, State, interstate, and local requirements" with regard to the control and abatement of air pollution.

TC

The Act requires the Environmental Protection Agency to establish National Ambient Air Quality Standards as necessary to protect public health, with an adequate margin of safety, from any known or anticipated adverse effects of a regulated pollutant (42 USC 7409). The Act also requires the establishment of national standards of performance for new or modified stationary sources of atmospheric pollutants (42 USC 7411) and requires specific emission increases to be evaluated so as to prevent a significant deterioration in air quality (42 USC 7470). Hazardous air pollutants, including radionuclides, are regulated separately (42 USC 7412). Air emissions are regulated by the Environmental Protection Agency in 40 CFR Parts 50 through 99. In particular, radionuclide emissions are regulated under the National Emission Standard for Hazardous Air Pollutants Program (see 40 CFR Part 61).

Safe Drinking Water Act, as amended [42 USC 300 (F) et seq.]

The primary objective of the Safe Drinking Water Act, as amended, is to protect the quality of the public water supplies and all sources of drinking water. The implementing regulations, administered by the Environmental Protection Agency unless delegated to the States, establish standards applicable to public water systems. They promulgate maximum contaminant levels (including those for radioactivity), in public water systems, which are defined as water systems that serve at least 15 service connections used by year-round residents or regularly serve at least 25 year-round residents. Safe Drinking Water Act requirements have been promulgated by the Environmental Protection Agency in 40 CFR Parts 100 through 149. For radioactive material, the regulations specify that the average annual concentration of manmade radionuclides in drinking water as delivered to the user by such a system shall not produce a dose equivalent to the total body or an internal organ greater than four mrem per year beta activity. Other programs established by the Safe Drinking Water Act include the Sole Source Aquifer Program, the Wellhead Protection Program, and the Underground Injection Control Program.

Clean Water Act, as amended (33 USC 1251 et seq.)

The Clean Water Act, which amended the Federal Water Pollution Control Act, was enacted to "restore and maintain the chemical, physical and biological integrity of the Nation's water." The Clean Water Act prohibits the "discharge of toxic pollutants in toxic amounts" to navigable waters of the United States. Section 313 of the Clean Water Act, as amended, requires all branches of the Federal Government engaged in any activity that might result in a discharge or runoff of pollutants to surface waters to comply with Federal, State, interstate, and local requirements.

In addition to setting water quality standards for the Nation's waterways, the Clean Water Act supplies guidelines and limitations for effluent discharges from point-source discharges and provides authority for the Environmental Protection Agency to implement the National Pollutant Discharge Elimination System permitting program. The National Pollutant Discharge Elimination System program is administered by the Water Management Division of the Environmental Protection Agency pursuant to regulations in 40 CFR Part 122 *et seq.* TC

Sections 401 and 405 of the Water Quality Act of 1987 added Section 402(p) to the Clean Water Act. Section 402(p) requires that the Environmental Protection Agency establish regulations for issuing permits for stormwater discharges associated with industrial activity. Although any stormwater discharge associated with industrial activity requires a National Pollutant Discharge Elimination System permit application, regulations implementing a separate stormwater permit application process have not yet been adopted by the Environmental Protection Agency.

Resource Conservation and Recovery Act, as amended (Solid Waste Disposal Act) (42 USC 6901 et seq.)

The treatment, storage, or disposal of hazardous and nonhazardous waste is regulated under the Solid Waste Disposal Act as amended by the Resource Conservation and Recovery Act and the Hazardous and Solid Waste Amendments of 1984. Pursuant to Section 3006 of the Act, any State that seeks to administer and enforce a hazardous waste program pursuant to the Resource Conservation and Recovery Act may apply for Environmental Protection Agency authorization of its program. The Environmental Protection Agency regulations implementing the Resource Conservation and Recovery Act are in 40 CFR Parts 260 through 280. These regulations define hazardous wastes and specify hazardous waste transportation, handling, treatment, storage, and disposal requirements.

The regulations imposed on a generator or a treatment, storage, or disposal facility vary according to the type and quantity of material or waste generated, treated, stored, or disposed of. The method of treatment, storage, or disposal also affects the extent and complexity of the requirements.

TC World events have resulted in significant changes in DOE's direction and operations. In particular, in April 1992 DOE announced the phase-out of reprocessing for the recovery of special nuclear materials. With these changes, DOE's focus has changed from reprocessing and recovery of materials to storage and ultimate disposition. This in turn has created uncertainty regarding the regulatory status of some nuclear materials in relation to the Resource Conservation and Recovery Act.

DOE has initiated discussion with the Environmental Protection Agency on the potential applicability of the Resource Conservation and Recovery Act to spent nuclear fuel. Further discussions with Environmental Protection Agency Headquarters and regional offices and State regulators are ongoing to develop a strategy for meeting any Resource Conservation and Recovery Act requirements that might apply.

Pollution Prevention Act of 1990

L7-1 The Pollution Prevention Act of 1990 establishes a national policy for waste management and pollution control that focuses first on source reduction, followed sequentially be environmentally safe recycling, treatment, and disposal. Disposal or releases to the environment should occur only as a last resort. In response, DOE has committed to participation in the Superfund Amendments and Reauthorization Act Section 313, U.S. Environmental Protection Agency 33/50 Pollution Prevention Program. The goal for facilities already involved in Section 313 compliance is to achieve by 1997 a 33-percent reduction in the release of 17 priority chemicals from a 1993 baseline. On August 3, 1993, President Clinton issued Executive Order 12856, expanding the 33/50 program such that DOE must reduce its total releases of all toxic chemicals by 50 percent by December 31, 1999. In addition, DOE is requiring each of its sites to establish site-specific goals to reduce the generation of all waste types.

Federal Facility Compliance Act (42 USC 6921 et seq.)

The Federal Facility Compliance Act, enacted on October 6, 1992, waives sovereign immunity for fines and penalties for Resource Conservation Recovery Act violations at Federal facilities. However, a provision postpones fines and penalties after 3 years for mixed waste storage prohibition violations at DOE sites and requires DOE to prepare plans for developing the required treatment capacity for mixed waste stored or generated at each facility. Each plan must be approved by the host State or the Environmental Protection Agency, after consultation with other affected States, and a consent order must be issued by the regulator requiring compliance with the plan. The Federal Facility Compliance Act further provides that DOE will not be subject to fines and penalties for land disposal restriction storage prohibition violations for mixed waste as long as it is in compliance with such an approved plan and consent order and meets all other applicable regulations. This would only apply to the nuclear materials discussed in this EIS if the Resource Conservation and Recovery Act would apply to storage and treatment of such materials. On September 20, 1995, the South Caroline Department of Health and Environmental Control approved with modification the Site Treatment Plan for SRS. DOE expects the signing of a consent order requiring compliance with the plan no later than October 6, 1995.

TC

National Historic Preservation Act, as amended (16 USC 470 et seq.)

The National Historic Preservation Act, as amended, provides that sites with significant national historic value be placed on the *National Register of Historic Places*. No permits or certifications are required under the Act. However, if a particular Federal activity could impact an historic property resource, consultation with the Advisory Council on Historic Preservation will usually generate a Memorandum of Agreement, including stipulations that must be followed to minimize adverse impacts. Coordination with the State Historic Preservation Officer ensures the proper identification of potentially significant sites and the implementation of appropriate mitigative actions.

Archaeological Resource Protection Act, as amended (16 USC 470aa et seq.)

This Act requires a permit for any excavation or removal of archaeological resources from public or Native American lands. Excavations must be undertaken for the purpose of furthering archaeological knowledge in the public interest, and resources removed are to remain the property of the United States. Consent must be obtained from the Indian Tribe owning lands on which a resource is located before a permit is issued, and the permit must contain terms or conditions requested by the Tribe.

Native American Grave Protection and Repatriation Act of 1990 (25 USC 3001)

This law directs the Secretary of Interior to assume responsibilities for repatriation of Federal archaeological collections and collections held by museums receiving Federal funding that are culturally affiliated with Native American Tribes. Major actions to be taken under this law include (1) establishing a review committee with monitoring and policy-making responsibilities, (2) developing regulations for repatriation, including procedures for identifying lineal descent or cultural affiliation needed for claims, (3) overseeing museum programs designed to meet the inventory requirements and deadlines of this law, and (4) developing procedures to handle unexpected discoveries of graves or grave goods during activities on Federal or tribal land.

American Indian Religious Freedom Act of 1978 (42 USC 1996)

This Act reaffirms Native American religious freedom under the First Amendment, and sets U.S. policy to protect and preserve the inherent and constitutional right of Native Americans to believe, express, and exercise their traditional religions. The Act requires that Federal actions avoid interfering with access to sacred locations and traditional resources that are integral to the practice of religion.

TC

Religious Freedom Restoration Act of 1993 (42 USC 2000bb et seq.)

This Act prohibits the Government, including Federal Departments, from substantially burdening the exercise of religion unless the government demonstrates a compelling governmental interest, and the action furthers a compelling Government interest and is the least restrictive means of furthering that interest.

Endangered Species Act, as amended (16 USC 1531 et seq.)

The Endangered Species Act, as amended, is intended to prevent the further decline of endangered and threatened species and to restore these species and their habitats. The Act is jointly administered by the United States Departments of Commerce and the Interior. Section 7 of the Act requires consultation with the U.S. Fish and Wildlife Service to determine if endangered and threatened species or their critical habitats are in the vicinity of the proposed action. Threatened or endangered species would not be affected by the activities associated with the alternatives considered in this EIS.

Migratory Bird Treaty Act, as amended (16 USC 703 et seq.)

The Migratory Bird Treaty Act, as amended, is intended to protect birds that have common migration patterns between the United States and Canada, Mexico, Japan, and Russia. It regulates the harvest of migratory birds by specifying things such as the mode of harvest, hunting seasons, and bag limits. The Act stipulates that it is unlawful at any time, by any means, or in any manner to "kill...any migratory bird." Although no permit for this project is required under the Act, DOE is required to consult with the U.S. Fish and Wildlife Service regarding impacts to migratory birds and to evaluate ways to avoid or minimize these effects in accordance with the U.S. Fish and Wildlife Service Mitigation Policy.

Bald and Golden Eagle Protection Act, as amended (16 USC 668-668d)

The Bald and Golden Eagle Protection Act makes it unlawful to take, pursue, molest, or disturb bald (American) and golden eagles, their nests, or their eggs anywhere in the United States (Sections 668, 668c). A permit must be obtained from the U.S. Department of the Interior to relocate a nest that interferes with resource development or recovery operations.

Wild and Scenic Rivers Act, as amended (16 USC 1271 et seq. 71:8301 et seq.) TC

The Wild and Scenic Rivers Act, as amended, protects certain selected rivers of the Nation that possess outstanding scenic, recreational, geological, fish and wildlife, historic, cultural, or other similar values. These rivers are to be preserved in a free-flowing condition to protect water quality and other vital national conservation purposes. The purpose of the Act is to institute a national wild and scenic rivers system, to designate the initial rivers that are a part of that system, and to develop standards for the addition of new rivers in the future.

Occupational Safety and Health Act of 1970, as amended (29 USC 651 et seq.)

The Occupational Safety and Health Act establishes standards to enhance safe and healthful working conditions in places of employment throughout the United States. The Act is administered and enforced by the Occupational Safety and Health Administration, a U.S. Department of Labor agency. While the Occupational Safety and Health Administration and Environmental Protection Agency both have a mandate to reduce exposures to toxic substances, the Occupational Safety and Health Administration's jurisdiction is limited to safety and health conditions that exist in the workplace environment. In general, under the Act, it is the duty of each employer to furnish all employees a place of employment free of recognized hazards likely to cause death or serious physical harm. Employees have a duty to comply with the occupational safety and health standards and all rules, regulations, and orders issued under the Act. The Occupational Safety and Health Administration regulations (29 CFR) establish specific standards telling employers what must be done to achieve a safe and healthful working environment. DOE places emphasis on compliance with these regulations at its facilities and prescribes through DOE Orders the Occupational Safety and Health Act standards that contractors shall meet, as applicable to their work at Government-owned, contractor-operated facilities (DOE Orders 5480.1B, 5483.1A). DOE keeps and makes available the various records of minor illnesses, injuries, and work-related deaths required by Occupational Safety and Health Administration regulations.

Noise Control Act of 1972, as amended (42 USC 4901 et seq.)

Section 4 of the Noise Control Act of 1972, as amended, directs all Federal agencies to carry out "to the fullest extent within their authority" programs within their jurisdictions in a manner that furthers a national policy of promoting an environment free from noise that jeopardizes health and welfare.

8.1.2 EXECUTIVE ORDERS

TC

Executive Order 11514 (Protection and Enhancement of Environmental Quality)

Executive Order 11514 requires Federal agencies to monitor and control their activities continually to protect and enhance the quality of the environment and to develop procedures to ensure the fullest practicable provision of timely public information and understanding of Federal plans and programs with environmental impact to obtain the views of interested parties. The DOE has issued regulations (10 CFR 1021) and DOE Order 5440.1E for compliance with this Executive Order.

Executive Order 11988 (Floodplain Management)

Executive Order 11988 requires Federal agencies to establish procedures to ensure that the potential effects of flood hazards and floodplain management are considered for any action undertaken in a floodplain and that floodplain impacts be avoided to the extent practicable.

Executive Order 11990 (Protection of Wetlands)

Executive Order 11990 requires Government agencies to avoid any short- and long-term adverse impacts on wetlands wherever there is a practicable alternative.

Executive Order 12856 (Right-to-Know Laws and Pollution Prevention Requirements)

Executive Order 12856 requires all Federal agencies to reduce the toxic chemicals entering any waste stream. This order also requires Federal agencies to report toxic chemicals entering waste streams; improve emergency planning, response, and accident notification; and encourage clean technologies and testing of innovative prevention technologies.

Executive Order 12898 (Environmental Justice)

Executive Order 12898 requires Federal agencies to identify and address disproportionately high and adverse human health or environmental effects of its programs, policies, and activities on minority and low-income populations.

8.1.3 DOE REGULATIONS AND ORDERS

TC

Through the authority of the Atomic Energy Act, DOE is responsible for establishing a comprehensive health, safety, and environmental program for its facilities. The regulatory mechanisms through which DOE manages its facilities are the promulgation of regulations and the issuance of DOE Orders.

The DOE regulations address such areas as energy conservation, administrative requirements and procedures, nuclear safety, and classified information. For the purposes of this EIS, relevant regulations include 10 CFR Part 834, Radiation Protection of the Public and the Environment; 10 CFR Part 835, Occupational Radiation Protection; 10 CFR Part 1021, Compliance with NEPA; and 10 CFR Part 1022, Compliance with Floodplains/Wetlands Environmental Review Requirements. DOE has enacted occupational radiation protection standards to protect DOE and its contractor employees. These standards are set forth in 10 CFR Part 83b, Occupational Radiation Protection; the rules in this part establish radiation protection standards, limits, and program requirements for protecting individuals from ionizing radiation resulting from the conduct of DOE activities, including those conducted by DOE contractors. The activity may be, but is not limited to, design, construction, or operation of DOE facilities. These regulations would be in effect for the construction and operation of any facilities associated with the management of foreign research reactor spent nuclear fuel.

DOE Orders generally set forth policy and the programs and internal procedures for implementing those policies. Table 8-1 lists the major DOE Orders pertaining to the eventual construction and operation of nuclear material facilities within the DOE Complex.

8.2 Emergency Management and Response

TC

8.2.1 AUTHORITIES AND DIRECTIVES

Emergency Planning and Community Right-to-Know Act of 1986 (42 USC 11001 et seq.) (also known as "SARA Title III")

The Emergency Planning and Community Right-to-Know Act of 1986 requires emergency planning and notice to communities and government agencies of the presence and release of specific chemicals. EPA implements this Act under regulations found at 40 CFR Parts 355, 370, and 372. Under Subtitle A of this Act, Federal facilities provide various information (such as inventories of specific chemicals used or stored and releases that occur from these facilities) to the State Emergency Response Commission and the Local Emergency Planning Committee to ensure that emergency plans are sufficient to respond to unplanned releases of hazardous substances. Implementation of the provisions of this Act began voluntarily in 1987, and inventory and annual emissions reporting began in 1988. In addition, DOE requires compliance with Title III as a matter of Departmental policy. The requirements for this Act were promulgated by EPA in 40 CFR Parts 350 through 372. The SRS submits hazardous chemical inventory reports to the South Carolina Department of Health and Environmental Control. The chemical inventory could change depending on the alternative(s) DOE implemented; however, subsequent reports would reflect any change to the inventory. L7-1

The Toxic Substances Control Act also regulates the treatment, storage, and disposal of certain toxic substances not regulated by the Resource Conservation and Recovery Act or other statutes, particularly polychlorinated biphenyls, chlorofluorocarbons, and asbestos.

Table 8-1. DOE Orders relevant to the Interim Management of Nuclear Materials.

DOE Order	Subject
1270.2B	Safeguards Agreement with the International Atomic Energy Agency (6-23-92)
1300.2A	Department of Energy Technical Standards Program (5-19-92)
1360.2B	Unclassified Computer Security Program (5-18-92)
1540.2	Hazardous Material Packaging for Transport-Administrative Procedures (9-30-86; Chg. 1, 12-19-88)
3790.1B	Federal Employee Occupational Safety and Health Program (1-7-93)
4330.4A	Maintenance Management Program (10-17-90)
4700.1	Project Management System (3-6-87)
5000.3B	Occurrence Reporting and Utilization of Operations Information (4-9-92)
5400.1	General Environmental Protection Program (11-9-88; Chg. 1, 6-29-90)
5400.2A	Environmental Compliance Issue Coordination (Errata 1-31-89)
5400.4	Comprehensive Environmental Response, Compensation, and Liability Act Requirements (10-6-89)
5400.5	Radiation Protection of the Public and the Environment (2-8-90; Chg. 2, 1-7-93)
5440.1E	National Environmental Policy Act Compliance Program (11-10-92)
5480.1B	Environmental, Safety and Health Program for DOE Operations (9-23-86; Chg. 2, 1-7-93)
5480.3	Environmental Requirements for the Packaging and Transportation of Hazardous Materials, Hazardous Substances, and Hazardous Wastes (7-9-85)
5480.4	Environmental Protection, Safety, and Health Protection Standards (5-15-84); Chg. 4, 1-7-93)
5480.6	Safety of Department of Energy-Owned Nuclear Reactors (9-23-86)
5480.7A	Fire Protection (2-17-93)
5480.8A	Contractor Occupational Medical Program (6-26-92)
5480.9	Construction Safety and Health Program (11-18-87)
5480.10	Contractor Industrial Hygiene Program (6-26-85)
5480.11	Radiation Protection for Occupational Workers (12-21-88; Chg. 2, 6-29-90)
5480.15	Department of Energy Laboratory Accreditation Program for Personnel Dosimetry (12-14-87)
5480.17	Site Safety Representatives (10-05-88)
5480.18A	Accreditation of Performance-Based Training for Category A Reactors and Nuclear Facilities (07-19-91)
5480.19	Conduct of Operations Requirements for DOE Facilities (7-9-90; Chg. 1, 5-18-92)
5480.20	Personnel Selection, Qualification, Training, and Staffing Requirements at DOE Reactor and Nonreactor Nuclear Facilities (2-20-91)
5480.21	Unreviewed Safety Questions (12-24-91)
5480.22	Technical Safety Requirements (2-25-92; Chg. 1, 9-15-92)
5480.23	Nuclear Safety Analysis Reports (4-10-92)
5480.24	Nuclear Criticality Safety (8-12-92)
5480.27	Equipment Qualification for Reactor and Nonreactor Nuclear Facilities (1-15-93)
5480.28	Natural Phenomena Hazards Mitigation (1-15-93)
5480.31	Startup and Restart of Nuclear Facilities (9-15-93)
5481.1B	Safety Analysis and Review System (9-23-86; Chg. 1, 5-19-87)
5482.1B	Environment, Safety, and Health Appraisal Program (9-23-86; Chg. 1, 11-18-91)
5483.1A	Occupational Safety and Health Program for DOE Contractor Employees at Government-Owned, Contractor-Operated Facilities (6-22-83)
5484.1	Environmental Protection, Safety, and Health Protection Information Reporting Requirements (2-21-81; Chg. 7, 10-17-90)
5500.1B	Emergency Management System (4-30-91; Chg. 1, 4-30-91)
5500.2B	Emergency Categories, Classes, and Notification and Reporting Requirements (4-30-91; Chg. 1, 2-27-92)
5500.3A	Planning and Preparedness for Operational Emergencies (6-8-92)
5500.4A	Public Affairs Policy and Planning Requirements for Emergencies (6-8-92)
5500.7B	Emergency Operating Records Protection Program (10-23-91)
5500.10	Emergency Readiness Assurance Program (4-30-91; Chg. 1, 2-27-92)
5530.3	Radiological Assistance Program (01-14-92; Chg. 1, 4-10-92)
5530.5	Federal Radiological Monitoring and Assessment Center (7-10-92)
5630.11B	Safeguards and Security Program (8-2-94)
5630.12A	Safeguards and Security Inspection and Evaluation Program (6-23-92)
5630.13A	Master Safeguards and Security Agreements (6-8-92)
5630.14A	Safeguards and Security Program Planning (6-9-92)
5630.15	Safeguards and Security Training Program (8-21-92)

TC

Table 8-1. (continued).

DOE Order	Subject
5630.16A	Safeguards and Security Acceptance and Validation Testing Program (6-3-93)
5630.17	Safeguards and Security (S&S) Standardization Program (9-29-92)
5631.6A	Personnel Security Assurance Program (9-15-92)
5632.1C	Protection and Control of Safeguards and Security Interests (7-15-94)
5633.3B	Control and Accountability of Nuclear Materials (9-7-94)
5634.1B	Facility Approval, Security Surveys, and Nuclear Materials Surveys (9-15-92)
5700.6C	Quality Assurance (8-21-91)
5820.2A	Radioactive Waste Management (9-26-88)
6430.1A	General Design Criteria (4-6-89)

Quantities of Radioactive Materials Requiring Consideration of the Need for an Emergency Plan for Responding to a Release (10 CFR Part 30.72 Schedule C)

This list is the basis for both the public and private sector to determine if the radiological materials they deal with must have an emergency response plan for unscheduled releases. It is one of the threshold criteria documents for DOE Hazards Assessments required by DOE Order 5500.3A, "Planning and Preparedness for Operational Emergencies." TC

Occupational Safety and Health Administration Emergency Response, Hazardous Waste Operations and Worker Right to Know (29 CFR)

This regulation sets down the Occupational Safety and Health Administration requirements for employee safety in a variety of working environments. It addresses employee emergency and fire prevention plans (Section 1910.38), hazardous waste operations and emergency response (Section 1910.120), and hazards communication (Section 1910.1200) that enables employees to be aware of the dangers they face from hazardous materials at their workplace.

Emergency Management and Assistance (44 CFR 1.1)

This regulation contains the policies and procedures for the Federal Emergency Management Act, National Flood Insurance Program, Federal Crime Insurance Program, Fire Prevention and Control Program, Disaster Assistance Program, and Preparedness Program including radiological planning and preparedness.

Hazardous Materials Tables & Communications, Emergency Response Information Requirements (49 CFR Part 172)

The regulatory requirements for marking, labeling, placarding, and documenting hazardous materials shipments are defined in this regulation. It also specifies the requirements for providing hazardous material information and training.

Public Law 93-288, as Amended by Public Law 100-707, "Robert T. Stafford Disaster Relief and Emergency Assistance Act," November 23, 1988

The Robert T. Stafford Disaster Relief and Emergency Assistance Act, Public Law 93-288, as amended, provides an orderly and continuing means of assistance by the Federal Government to state and local governments in carrying out their responsibilities to alleviate the suffering and damage resulting from disasters. The President, in response to a state Governor's request, may declare an "emergency" or "major disaster" to provide Federal assistance under the Act. The President, in Executive Order 12148, delegated all functions, except those in Sections 301, 401, and 409, to the Director, Federal Emergency Management Agency. The Act provides for the appointment of a Federal Coordinating Officer who will operate in the designated area with a State Coordinating Officer for the purpose of coordinating State and local disaster assistance efforts with those of the Federal Government. TC

Public Law 96-510, "Comprehensive Environmental Response, Compensation, and Liability Act of 1980," Section 104(i), 42 U.S.C 9604(i)

More popularly known as "Superfund," this Act provides the needed general authority for Federal and state governments to respond directly to hazardous substances incidents. The Act requires reporting of spills, including radioactive, to the National Response Center.

Public Law 98-473, Justice Assistance Act of 1984

These Department of Justice regulations implement the Emergency Federal Law Enforcement Assistance functions vested in the Attorney General. Those functions were established to assist state and local units of government in responding to a law enforcement emergency. The Act defines the term "law enforcement emergency" as an uncommon situation that requires law enforcement, that is or threatens to become of serious or epidemic proportions, and with respect to which state and local resources are inadequate to protect the lives and property of citizens, or to enforce the criminal law.

Emergencies that are not of an ongoing or chronic nature, such as the Mount Saint Helens volcanic eruption, are eligible for Federal law enforcement assistance. Such assistance is defined as funds, equipment, training, intelligence information, and personnel. Requests for assistance must be submitted in writing to the Attorney General by the chief executive office of a state. The Plan does not cover the provision of law enforcement assistance. Such assistance will be provided in accordance with the regulations referred to in this paragraph [28 CFR Part 65, implementing the Justice Assistance Act of 1984] or pursuant to any other applicable authority of the Department of Justice.

Communications Act of 1934, as Amended

This Act gives the Federal Communications Commission emergency authority to grant Special Temporary Authority on an expedited basis to operate radio frequency devices.

8.2.2 EXECUTIVE ORDERS

Executive Order 10480, as Amended, "Further Providing for the Administration of the Defense Mobilization Program," August 1953

TC

Part II of the Order delegates to the Director, Federal Emergency Management Agency, with authority to redelegate, the priorities and allocation functions conferred on the President by Title I of the Defense Production Act of 1950, as amended.

Executive Order 12148, "Federal Emergency Management," July 20, 1979

Executive Order 12148 transferred functions and responsibilities associated with Federal emergency management to the Director, Federal Emergency Management Agency. The Order assigns the Director, Federal Emergency Management Agency, the responsibility to establish Federal policies for and to coordinate all civil defense and civil emergency planning, management, mitigation, and assistance functions of Executive Agencies.

Executive Order 12472, "Assignment of National Security and Emergency Preparedness Telecommunications Functions," April 3, 1984

Executive Order 12472 establishes the National Communication System, which consists of the telecommunications assets of the organizations represented on the National Communication System

Committee of Principals and an administrative structure consisting of the Executive Agent, the National Communication System Committee of Principals, and the Manager. The Committee of Principals consists of representatives from Federal departments, agencies, or entities, designated by the President, which lease or own telecommunications facilities or services of significance to national security or emergency preparedness.

Executive Order 12656, "Assignment of Emergency Preparedness Responsibilities," November 1988

This order assigns emergency preparedness responsibilities to Federal departments and agencies.

8.2.3 EMERGENCY PLANNING DOCUMENTS

"Federal Radiological Emergency Response Plan," November 1985

This document is to be used by Federal agencies in peacetime radiological emergencies. It primarily concerns the offsite Federal response in support of state and local governments with jurisdiction for the emergency. The Federal Radiological Emergency Response Plan provides the Federal Government's concept of operations based on specific authorities for responding to radiological emergencies, outlines Federal policies and planning assumptions that underlie this concept of operations and on which Federal agency response plans were based, and specifies authorities and responsibilities of each Federal agency that might have a significant role in such emergencies. TC

"National Plan for Telecommunications Support [in Non-Wartime Emergencies]," January 1992

This plan provides guidance in planning for and providing telecommunications support for Federal agencies involved in emergencies, major disasters, and other urgent events, excluding war.

Department of Defense Directive 3025.1, "Military Support to Civil Authorities," 1992

This directive outlines Department of Defense (DOD) policy on assistance to the civilian sector during disasters and other emergencies. Use of DOD military resources in civil emergency relief operations will be limited to those resources not immediately required for the execution of the primary defense mission. Normally, DOD military resources will be committed as a supplement to non-DOD resources that are required to cope with the humanitarian and property protection requirement caused by the emergency. In any emergency, commanders are authorized to employ DOD resources to save lives, prevent human suffering, or mitigate great property loss. On declaration of a major disaster under the provisions of Public Law 93-288, as amended, the Secretary of the Army is the DOD Executive Agent, and the Director of Military Support is the action agent for civil emergency relief operations. Military personnel will be under command of and directly responsible to their military superiors and will not be used to enforce or execute civil law in violation of 18 USC 1385, except as otherwise authorized by law. Military resources shall not be procured, stockpiled, or developed solely to provide assistance to civil authorities during emergencies.

Federal Preparedness Circular 8, "Public Affairs in Emergencies"

This Circular establishes the Interagency Committee on Public Affairs in Emergencies to coordinate public information planning and operations for management of emergency information. The Circular was reviewed in draft by the Interagency Committee on Public Affairs in Emergencies and will receive formal department and agency review.

TC *American Red Cross Disaster Services Regulations and Procedures, ARC 3003, January 1984*

This document details the delegation of disaster services program responsibilities to officials and units of the American Red Cross. It also defines are the American Red Cross administrative regulations and procedures for disaster planning, preparedness, and response.

Statement of Understanding Between the Federal Emergency Management Agency and the American National Red Cross, January 22, 1982

The statement of understanding between the Federal Emergency Management Act and the American National Red Cross describes major responsibilities in disaster preparedness planning and operations in the event of a war-caused national emergency or a peacetime disaster, outlines areas of mutual support and cooperation, and provides a frame of reference for similar cooperative agreements between State and local governments and the operations headquarters and chapters of the American Red Cross.

APPENDIX A. RADIOACTIVE MATERIALS AT THE SAVANNAH RIVER SITE

This appendix provides:

- General information on vulnerabilities associated with nuclear materials at the Savannah River Site
- A list of nuclear materials stored at the SRS, grouped into three general categories: (1) Stable, (2) Programmatic, and (3) Candidates for Stabilization (as described in Chapter 1)
- Information on the quantities of weapons-usable material (plutonium and highly enriched uranium) in nuclear materials at the SRS, along with the mass of irradiated fuels and targets

A.1 Material/Packaging/Storage Vulnerabilities

A.1.1 SOLUTIONS

The following significant vulnerabilities could develop during extended solution storage: L11-6 L11-14 L11-16

- Leaks or spills that result in releases of radioactive material
- Increased difficulty in maintaining solution chemistry to control such parameters as corrosivity
- Possible precipitation and accumulation of fissile material into configurations that could cause criticality events.

The potential consequences from solution storage vulnerabilities are highest to workers who work close to possible leaks, contamination, and criticality events.

A.1.2 PLUTONIUM/URANIUM OXIDE SCRAP AND RESIDUES

Many forms of scrap and residues are corrosive, chemically reactive, and difficult to contain, particularly when they are exposed to air and moisture due to poor package design or from packaging failure stemming from radiolysis and pressure buildup. Scrap and residues in contact with plastics can cause radiolysis, hydrogen generation, and pressurization, making these packages susceptible to leaks or ruptures. The instability and chemical reactivity of scrap and residues have already caused packaging failures at some DOE facilities.

The large quantity of scrap and residues, the diversity of packaging, and ongoing package degradation create the potential for accidental plutonium releases and worker exposures. Workers face the highest potential consequences due to their proximity to failed scrap and residue packages during storage and handling activities. The repackaging activities proposed in this EIS would alleviate these vulnerabilities.

A.1.3 PLUTONIUM METALS

The most significant vulnerabilities from the storage of plutonium metal stem from oxidation and radiolysis. Current packaging methods can allow air and moisture to enter, resulting in oxidation. When a container of plutonium metal also contains plastic bags or a food pack can with plastic seals, the plastic degrades. This might happen within a year depending on the type of plastic and its proximity to the plutonium metal. Chemical and radiolytic reactions between plastic and plutonium can cause inner packaging to leak, allowing air to enter and consequent oxidation of plutonium and formation of hydride. Plutonium oxide can swell to a volume approximately seven times that of the original plutonium metal and rupture multiple barrier containers. Even in the absence of plastic, plutonium metal will eventually oxidize if air or moisture enters. As a result, containers that are not completely sealed are not suitable for extended storage. The repackaging activities proposed in this EIS would alleviate these vulnerabilities.

L11-6
L11-14
L11-16

A.1.4 PLUTONIUM OXIDES

The most significant vulnerabilities associated with the storage of plutonium oxide stem from the oxide form, radiolysis, and chemical reactions. In general plutonium oxide is a fine powder, like talcum powder, and is respirable, thus posing a severe hazard to workers not wearing respiratory protection. The powdery oxide can escape from ruptured packages. Inhalation of very small (microgram) quantities of plutonium oxide can result in significant radiation doses. Oxide stored in plastic packaging can result in failures similar to those associated with metal. Hydrogen generated from radiolysis and chemical reactions with plastic or absorbed moisture can cause gas buildup and contribute to plutonium releases. The repackaging activities proposed in this EIS would alleviate these vulnerabilities.

A.1.5 SPENT FUEL AND TARGETS IN WET STORAGE

The aluminum-clad fuel and targets (e.g., Mark-31s and Mark-16/22s) have exhibited corrosion of the cladding due to the elevated conductivity level of the water in the reactor disassembly basin and galvanic action between dissimilar metals such as the aluminum cladding on fuel and the stainless steel of the fuel storage racks. The corrosion has resulted in a release of fission products and uranium and plutonium oxide to the basin water. These create vulnerabilities by increasing basin radioactivity levels. L11-6 L11-14 L11-16

A.2 SRS Nuclear Materials

DOE has evaluated the nuclear materials stored at the Savannah River Site and grouped them into three general categories: (1) Stable, (2) Programmatic, and (3) Candidates for Stabilization. Table A-1 lists the materials grouped in these categories and briefly describes each material and its storage location.

Table A-1. Savannah River Site nuclear materials.[a]

STABLE MATERIAL

Spent nuclear fuels - Approximately 3,000 uranium-plutonium fuel elements from a number of reactors around the world, clad with aluminum, stainless steel, zirconium, hastaloy, or nichrome are stored in the Receiving Basin for Offsite Fuel (RBOF). This section on RBOF spent nuclear fuel has been divided into aluminum-based fuels and other (nonaluminum) spent fuels.

		Fuel/target in inventory		Storage activities	
Source of fuel/target	Composition	Number	Units	Number of storage positions	Storage units
Aluminum-based spent nuclear fuels					
Mark-22	HEU-Al alloy, Al-clad	2	Tube	2	Tube
Mark-16	HEU-Al alloy, Al-clad	2	Tube	2	Tube
Mark-14	HEU-Al alloy, Al-clad	1	Can	1	Can
Argonne National Laboratory Janus Reactor	HEU-Al alloy, Al-clad	19	Assemblies	4	Bundles
Advanced Thermal Source Reactor	HEU-Al alloy, Al-clad	21	Assemblies	4	Bundles
Massachusetts Institute of Technology Reactor	HEU-Al alloy, Al-clad	56	Assemblies	19	Bundles
Missouri University Research Reactor	HEU-Al alloy, Al-clad	112	Assemblies	56	Bundles
Rhode Island Nuclear Center	HEU-Al alloy, Al-clad	70	Assemblies	13	Bundles
University of Michigan Reactor	MEU-Al alloy, Al-clad	48	Assemblies	8	Bundles
University of Virginia Reactor	HEU-Al alloy, Al-clad	44	Assemblies	8	Bundles
Nereide (French) Research Reactor	MEU-Al alloy, Al-clad	46	Assemblies	8	Bundles

TC

Table A-1. (continued).

		Fuel/target in inventory		Storage activities	
Source of fuel/target	Composition	Number	Units	Number of storage positions	Storage units
Japanese Material Test Reactor	HEU-Al alloy, Al-clad	71	Assemblies	15	Bundles
French Hot Flux Research Reactor	HEU-Al alloy, Al-clad	4	Assemblies	4	Assemblies
Oak Ridge Research Reactor	MEU-Al alloy, Al-clad	65	Assemblies	12	Bundles
	HEU-Al alloy, Al-clad	100	Assemblies	17	Bundles
Sterling Forest	HEU oxide, Al-clad	678	Cans	226	Bundles
	HEU-Al alloy, Al-clad	200	Assemblies	34	Bundles
Taiwan Research Reactor	NU with Pu-239, Al-clad	62	Cans	62	Cans
Experimental Breeder Reactor-II	DU metal, Al-clad	59	Cans	59	Cans
Special curium targets		63	Slugs	63	Slugs
Mark-18 special Am-241 target		65	Slugs	65	Slugs
Other		51	Slugs	51	Slugs
<u>Other spent nuclear fuels</u>					
Carolinas-Virginia Tube	U oxide, Zr- or SS-clad	34	Rods	1	Can
Dresden	U oxide-Th oxide, SS-clad (LEU)		Rods	4	Sleeves
	U oxide-Th oxide, SS-clad (MEU)		Rods	1	Sleeves
	U oxide-Th oxide, SS-clad (HEU)		Rods	23	Sleeves
	U oxide-Th oxide, SS-clad (LEU)		Scrap	2	Cans
Elk River reactor	U oxide-Th oxide, SS-clad	189	Rods	38	Bundles
LWR samples	U oxide-Pu oxide, Zr- or SS-clad		Pieces	5	Cans
H. B. Robinson	U oxide-Pu oxide, Zr-clad in SS casing	4	Pieces	1	Can
Saxton	U oxide-Pu oxide, SS clad	567	Rods	8	Cans
	U oxide-Pu oxide, SS-clad	64	Rods	1	Can
	U oxide-Pu oxide, SS-clad	2	Assemblies	2	Assemblies
	U oxide-Pu oxide, SS-clad		Pieces	2	Cans
	U oxide, Zr-clad		Pins	4	Cans
	U oxide, Zr-clad	2	Tubes	1	Can
Vallecitos Boiling Water Reactor	U oxide, Zr-clad (LEU)		Assemblies	1	Bundle
	U oxide, Zr-clad (MEU)		Assemblies	1	Bundle
	U oxide, Zr-clad (HEU)		Assemblies	1	Bundle

TC

Table A-1. (continued).

		Fuel/target in inventory		Storage activities	
Source of fuel/target	Composition	Number	Units	Number of storage positions	Storage units
Babcock and Wilcox scrap	U oxide-Pu oxide, SS-clad		Scrap	1	Can
Experimental Breeder Reactor-II	U oxide, Zr- or SS-clad	8	Rods	1	Can
	U oxide, Zr- or SS-clad		Segments	1	Can
Experimental Boiling Water Reactor	U oxide, Zr- or SS-clad			4	Cans
	U oxide, Zr- or SS-clad (MEU)			1 4	Can Bundles
	U oxide-Zr oxide, Zr-clad (LEU)	114	Assemblies	114	Assemblies
	U oxide-Zr oxide, Zr-clad (NU)	34	Assemblies	34	Assemblies
	U oxide-Zr oxide, Zr-clad (DU)	22	Assemblies	22	Assemblies
	U oxide-Zr oxide-Ca oxide, Zr-clad	32	Assemblies	32	Assemblies
	U oxide-Pu oxide, Zr-clad	20	Assemblies	20	Assemblies
EPR-1 Gas Cooled Reactor Experiment	Pu oxide, SS-clad		Pieces	1	Can
	U oxide or U oxide-Be oxide, Hastelloy-clad		Scrap	4	Cans
	U oxide or U oxide-Be oxide, Hastelloy-clad		Assemblies	2	Cans
	U oxide or U oxide-Be oxide, Hastelloy-clad	66	Pins	66	Pins
Heavy Water Components Test Reactor	U or U oxide, Zr-clad (LEU)	3	Tubes	3	Tubes
	U or U oxide, Zr-clad (LEU)		Scrap	1	Can
	U or U oxide, Zr-clad (LEU)	5	Assemblies	5	Assemblies
	U or U oxide, Zr-clad (LEU)	2	Cans	2	Cans
	U, Zr-clad	3	Slugs	3	Slugs
	U or U oxide, Zr-clad (LEU)	35	Cans	35	Cans
	U or U oxide, Zr-clad (DU)	10	Tubes	10	Tubes
	U or U oxide, Zr-clad (DU)	9	Bundles	9	Bundles
	U oxide, Zr-clad (DU)	8	Assemblies	8	Assemblies
	U, Zr-clad (DU)	4	Bundles	4	Bundles
	U or U oxide, Zr-clad (DU)	5	Slugs	5	Slugs
	U or U oxide, Zr-clad (DU)	4	Cans	4	Cans
	U-Zr, Zr-clad	45	Assemblies	45	Assemblies
	U-Zr, Zr-clad	19	Cans	19	Cans
	U-Th, Zr-clad	1	Assemblies	1	Assemblies
High Temperature Reactor Experiment	U oxide-Be oxide, nichrome-clad		Pieces	13	Cans
Mobile Low Power Plant No. 1	U oxide and Pu oxide-Be oxide, SS-clad	68	Assemblies	68	Assemblies
Oak Ridge National Laboratory S1W-1 rods	U, Zr-clad		Rods	3	Cans
Oak Ridge National Laboratory Mixed Oxide	U oxide-Pu oxide, Zr- or SS-clad	1	Can	1	Can
Shippingport	U oxide-Pu oxide, Zr-clad	1	Can	1	Can

TC

Table A-1. (continued).

Source of fuel/target	Composition	Fuel/target in inventory		Storage activities	
		Number	Units	Number of storage positions	Storage units
Special Power Excursion Reactor Test-3	U oxide, Zr-clad	3	Cans	3	Cans
Sodium Reactor Experiment	U oxide-Th oxide, SS-clad, HEU	36	Cans	36	Cans
	UC, SS-clad	1	Can	1	Can
Canadian Deuterium Uranium Reactor	U oxide, Zr-clad		Rods	3	Cans
	U oxide, Zr-clad		Pieces	3	Cans

Description and storage management activities	Location
Research and development material - About 260 nuclear materials, used in routine laboratory research and development activities. When not in use these materials are packaged in cans, bottles, or sample carriers and stored in laboratory hoods, gloveboxes, or cells:	
Americium-241 oxide scrap from Savannah River Laboratory test work	SRTC
Americium, curium, plutonium-238 solution	SRTC
Depleted uranium metal rods for hydride development	SRTC
Depleted uranium nitrate crystals	SRTC
Depleted uranium oxide and ring sections from tubes	SRTC
Depleted uranium oxide-aluminum powder compacted	SRTC
Depleted uranium scrap	SRTC
Depleted uranium slurry	SRTC
Enriched uranium floor sweepings	SRTC
Liquid samples from Old FB-Line ductwork (americium, curium, and plutonium-238)	SRTC
Liquid samples from Old HB-Line ductwork	SRTC
Mark-16 enriched uranium oxide powder metallurgy tube	SRTC
Natural uranium gel sphere samples	SRTC
Neptunium solution samples	SRTC
Plutonium oxide and anode heel residues	SRTC
Thorium oxide	SRTC
Unirradiated natural uranium	Building 772-F
Unirradiated normal uranium for research and development	SRTC
Uranium-233 oxide from Oak Ridge	Building 772-F
Uranyl nitrate solution sample	SRTC
Special actinides - Two thorium oxide spheres in Building 235-F that DOE used as production guides for startup of the Plutonium Fuel Fabrication Facility in 1977; four containers of neptunium scrap in HB-Line.	Building 235, HB-Line
Depleted uranium solutions - Approximately 276,000 liters (73,000 gallons) of depleted uranium solution in two stainless-steel tanks in F-Canyon, seven stainless-steel tanks in A-Line, and one stainless-steel TNX tank truck.	
Depleted uranium solution - TNX Tank Truck	F-Area Outside Facility
Depleted uranium solutions	F-Canyon, F-Area Outside Facility

TC

Table A-1. (continued).

Description and storage management activities	Location
Unirradiated uranium in M-Area - More than 315,000 items consisting of uranium and lithium residues from fabrication of fuel and targets for the reactors (mostly unirradiated Mark-31 targets in various stages of fabrication). Uranium is mostly depleted but a small amount of fully enriched uranium is in storage. Lithium stocks are lithium metal or as lithium-aluminum alloy. These materials are stored dry and routinely monitored and inventoried. If corrective actions are needed, the material would be repackaged:	
Bare Mark-25A cores and bare Mark-25B cores	Building 313-M
Canned Mark-31 slugs	Building 313-A
Canned Mark-31 slugs, depleted uranium, nickel-plated and aluminum-clad	Building 313-M
Depleted uranium Mark-31 scrap, no cladding (reject cores)	Building 313-M
Depleted uranium sludge	Building 322-M
Depleted uranium sludge	Building 341-1M
Enriched lithium metal in cans	Building 320-M
Enriched uranium oxide in filter cake	Building 313-M
Lithium-aluminum alloy control rods and sparger slugs	Building 315-M
Lithium-aluminum alloy in castings, billets, and cores	Building 315-M
Lithium-aluminum control rods, spargers, and targets	Building 315-M
Mark-15B canned slugs	Building 313-M
Mark-25 depleted uranium dummy core	Building 313-M
Mark-31 depleted uranium fuel with aluminum cladding	Building 313-M
Natural lithium metal in cans	Building 320-M
Unclad normal uranium metal fuel pins	Building 313-M
Unirradiated Mark-15A cores	Building 313-M
Reactor components - Approximately 7,500 items stored dry in reactor assembly areas. Materials are unirradiated and consist of various reactor components and canned fabrication components and scrap. Included are control rods, spargers, and targets consisting of lithium-aluminum alloy clad in aluminum. Also included are aluminum-clad enriched uranium-aluminum fuel tubes and canned pieces and scraps.	
Aluminum-enriched uranium alloy, aluminum-clad slugs from Savannah River Site Nuclear Test Gauge	K-Reactor Assembly
Enriched uranium grinding residues from Building 321-M	K-Reactor Assembly
Enriched uranium slugs, aluminum-clad, from Building 321-M Nuclear Test Gauge	K-Reactor Assembly
Enriched uranium-aluminum alloy Mark-16 and Mark-22 tubes, scrap, standards	K-Reactor Assembly
Enriched uranium-aluminum floor sweepings	K-Reactor Assembly
Mark-22 fuel tubes, enriched uranium with aluminum cladding	K-Reactor Assembly
Unirradiated Mark-16B assemblies, spares for reactor charge	K-Reactor Assembly
Uranium-aluminum fuel tube ring section	K-Reactor Assembly
Uranium-aluminum grinding fines from fuel tube grinding	K-Reactor Assembly
Lithium-aluminum control rods, spargers, and targets	K- and L-Reactor Assembly
Unirradiated contaminated lithium aluminum targets	K- and L-Reactor Assembly
Unirradiated Mark-16B assemblies, spares for reactor charge	L-Reactor Assembly
Unirradiated Mark-22 assemblies with lithium target tubes	K-Reactor Assembly
Unirradiated Mark-22 fuel assemblies (enriched uranium)	L-Reactor Assembly
Depleted uranium oxide - Approximately 36,000 208-liter (55-gallon) drums containing approximately 20 metric tons (22 tons) of uranium. The uranium-235 concentration is mostly below naturally occurring uranium. These drums of uranium oxide are stored in buildings.	R-Reactor Assembly, Buildings 221-21F, 221-22F, 704-R, 714-7N, 728-F, 730-F, 772-7B

TC

Table A-1. (continued).

Description and storage management activities	Location
Uranyl nitrate solution - Two stainless-steel tanks outside the TNX facility contain approximately 17,400 liters (4,600 gallons) of depleted uranium nitrate solution.	TNX
Sources, standards, and samples - SRS uses sources and standards in its many monitoring and analytical functions. Most of these sources and standards contain a small amount of nuclear material. DOE estimates that more than 20,000 sources and standards are in use.	Sitewide
PROGRAMMATIC MATERIALS	
Plutonium-242	
Solution - Approximately 13,200 liters (3,500 gallons) of nitrate solution high in plutonium-242, stored in a single stainless-steel tank.	H-Canyon
Americium and Curium	
Solution - Approximately 14,000 liters (3,800 gallons) of americium-243 and curium-244 nitrate solutions are stored in a single stainless-steel tank.	F-Canyon
Solids - Sixty-five Mark-18 targets stored in the Receiving Basin for Offsite Fuel, 114 slugs stored in RBOF, and 60 slugs stored in the P-Reactor disassembly basin.	RBOF P-Reactor Disassembly Basin
Neptunium-237	
Solutions - Approximately 6,100 liters (1,600 gallons) of neptunium nitrate solutions stored in two stainless-steel tanks.	H-Canyon
Targets - Nine Mark-53 unirradiated neptunium-aluminum alloy targets clad with aluminum, stored dry in borated storage racks.	Building 321-M
CANDIDATE MATERIALS FOR STABILIZATION	
Plutonium-239 solutions - Approximately 34,000 liters (9,000 gallons) of plutonium nitrate solutions stored in stainless-steel tanks.	H-Canyon
Highly enriched uranium solutions - Approximately 228,000 liters (60,000 gallons) of enriched uranium (approximately 60 percent uranium-235) nitrate solution. Solution is in two canyon tanks and five outside tanks. All tanks are stainless-steel and outside tanks are in concrete dikes large enough to contain the solution volume of the largest single tank.	H-Canyon, H-Area Outside Facilities
Plutonium-bearing vault material - Approximately 3,000 packages of material. The material contains alloys, compounds, oxides, large metal pieces such as buttons and ingots, and metal fragments, and consists predominantly of plutonium-239 with some uranium-235. In addition, some scrap containing predominately plutonium-238 material is stored in various locations.	
Low-uranium plutonium solids - Approximately 1,600 packages of plutonium-bearing solids containing low enough concentrations of uranium-235 to be processable in F-Area. Material is packaged in a metal can in a plastic bag in another metal pail or can (can/bag/can configuration), stored in a vault or glovebox.	
Plutonium solids - Approximately 1,000 packages containing more than 100 grams (3.5 ounces) of fissile material in a container. They include alloys, metals, compounds, oxides, and large metal pieces (e.g., buttons and ingots) of plutonium-239 with minimal other actinide impurities other than americium-241, the decay daughter of plutonium-241:	
Depleted uranium-plutonium alloy from Argonne	Building 235-F
Depleted uranium-plutonium alloy from Zero Power Plutonium Reactor	Building 235-F

TC

Table A-1. (continued).

Description and storage management activities	Location
High-fired plutonium oxides from Rocky Flats	Building 235-F
Impure plutonium metal from Livermore	Building 235-F
Mixed plutonium-uranium oxide from Oak Ridge	Building 235-F
Natural uranium compounds from Battelle and Argonne	FB-Line
Natural uranium-plutonium alloy from Argonne	Building 235-F
Plutonium finished product	FB-Line
Plutonium metal	Building 235-F
Plutonium metal (Category 3) from Hanford	FB-Line
Plutonium metal	FB-Line
Plutonium metal from Argonne	FB-Line
Plutonium metal from Livermore	Building 235-F
Plutonium metal from Los Alamos	FB-Line
Plutonium oxide from Argonne	FB-Line
Plutonium oxide from Hanford	FB-Line
Plutonium oxide from Livermore	FB-Line
Plutonium oxide from Nuclear Fuel Services	FB-Line
Plutonium oxide from Rocky Flats	FB-Line
Plutonium-americium oxide	FB-Line
Plutonium-americium oxides from Rockwell	FB-Line
Plutonium-bearing alloy from Hanford	FB-Line
Plutonium-depleted uranium alloy from Argonne	FB-Line
Plutonium-depleted uranium compounds from Argonne	FB-Line
Plutonium-depleted uranium compounds from Hanford	FB-Line
Plutonium-depleted uranium compounds from Hanford and Argonne	FB-Line
Plutonium-depleted uranium oxide from Hanford	FB-Line
Plutonium-depleted uranium oxide material from Argonne	FB-Line
Plutonium-depleted uranium-molybdenum alloy (Zero Power Plutonium Reactor)	FB-Line
Plutonium-natural uranium compounds from Argonne	Building 235-F
Plutonium-natural uranium compounds from Argonne and Hanford	FB-Line
Plutonium-natural uranium oxide from Hanford	FB-Line
Plutonium-natural uranium oxides (high-fired) from Hanford	FB-Line
Plutonium-natural uranium oxides from Hanford	FB-Line
Scrap and residue plutonium - Approximately 600 packages containing reactive or unknown plutonium forms with unknown reactivity such as plutonium turnings, sand, slag, crucibles, some plutonium compounds and metal fragments, and other alloys, metals, compounds, and oxides of plutonium-239 having minimal other actinide impurities. Sand, slag, and crucibles are a process residue containing potentially reactive calcium and fluorides.	
Analytical laboratory sample residues containing plutonium-242 oxide	Building 772-F
Anode heel metal (americium-241 and plutonium-239) from Rocky Flats	FB-Line
Depleted uranium oxide material from Battelle	Building 235-F
Depleted uranium-plutonium pellets and powder	SRTC
FB-Line cabinet floor sweepings (plutonium)	FB-Line
Formed plutonium metal from Livermore	FB-Line
Miscellaneous plutonium from crucibles	FB-Line
Natural uranium compounds from Battelle and Argonne	FB-Line
Natural uranium-plutonium oxides (low-fired) from Battelle	Building 235-F
Plutonium and natural uranium-depleted uranium pellets	FB-Line

TC

Table A-1. (continued).

Description and storage management activities	Location
Plutonium and sweepings received from Los Alamos	FB-Line
Plutonium compounds from Westinghouse Electric	FB-Line
Plutonium metal alloy and graphite residues from Rocky Flats	Building 235-F
Plutonium metal (formed) from Livermore	FB-Line
Plutonium metal from Los Alamos (test dissolution)	Building 235-F
Plutonium metal pieces	FB-Line
Plutonium metal button fragments	FB-Line
Plutonium metal turnings	FB-Line
Plutonium metal turnings from Rocky Flats	FB-Line
Plutonium oxide	FB-Line
Plutonium oxide from Hanford	FB-Line
Plutonium oxide in crucible from Fast Flux Test Reactor at Hanford	FB-Line
Plutonium powder	FB-Line
Plutonium residues (sand, slag, and crucible)	FB-Line
Plutonium rods	FB-Line
Plutonium scrub alloy or salt buttons from Rocky Flats	Building 235-F
Plutonium turnings	FB-Line
Plutonium-depleted uranium and plutonium-depleted uranium-silicon from Argonne	FB-Line
Plutonium-depleted uranium and plutonium-natural uranium compounds from Nuclear Energy	FB-Line
Plutonium-depleted uranium material from Argonne	FB-Line
Plutonium-depleted uranium material from Battelle	FB-Line
TC Plutonium-depleted uranium material	FB-Line
Plutonium-depleted uranium oxide from Battelle	Building 235-F
Plutonium-depleted uranium residue from Hanford	FB-Line
Plutonium-depleted uranium residue from Oak Ridge	FB-Line
Plutonium-depleted uranium residue from West Virginia Medical Center	FB-Line
Plutonium-natural uranium compounds from Argonne	FB-Line
Plutonium-natural uranium compounds from Battelle	Building 235-F
Plutonium-natural uranium oxides	FB-Line
Plutonium-oxide high in plutonium-240	FB-Line
Plutonium-zirconium alloy from Argonne	FB-Line
Pump oxide mix from Hanford and Oak Ridge	FB-Line
Sand, slag, and crucible residues from Rockwell	FB-Line
Scrap depleted uranium-plutonium oxide fuel rods from Savannah River Laboratory	Building 235-F

<u>Mixed solids</u> - This grouping consists of approximately 500 packages of plutonium or neptunium alloys, metals, compounds, and oxides mixed with enriched uranium. Package configuration is can/bag/can or bag/can/bag/can, stored in vaults.

Fissile mixed solids - Approximately 300 packages containing more than 100 grams (3.5 ounces) of fissile material per package:

Enriched uranium alloy (passivated) from Argonne	Building 235-F
Enriched uranium alloy solids and powder from Los Alamos	Building 235-F
Enriched uranium metal or oxide from Oak Ridge	Building 235-F
Enriched uranium oxide (high-fired and contaminated with plutonium)	Building 235-F
Enriched uranium oxide (high-fired with possible plutonium contamination) from Westinghouse	Building 235-F

Table A-1. (continued).

Description and storage management activities	Location
Enriched uranium oxide contaminated with plutonium from Rocky Flats	Building 235-F
Enriched uranium oxide from Rocky Flats	Building 235-F
Enriched uranium parts (plutonium contaminated) from Livermore	Building 235-F
Enriched uranium-plutonium alloy from Argonne	FB-Line
Enriched uranium-plutonium compound from Argonne	Building 235-F, FB-Line
Enriched uranium-plutonium compound from Rocky Flats	235-F
Enriched uranium-plutonium compound from West Virginia University reactor	235-F
Enriched uranium-plutonium compound from Westinghouse	FB-Line
Enriched uranium-plutonium compounds from Battelle	Building 235-F, FB-Line
Enriched uranium-plutonium high-fired oxides from Los Alamos	Building 235-F
Enriched uranium-plutonium metal and powder from Battelle	Building 235-F
Enriched uranium-plutonium oxide (high-fired) from Atomics International	Building 235-F
Enriched uranium-plutonium oxide from Battelle	Building 235-F
Enriched uranium-plutonium oxide from Rocky Flats	Building 235-F
Enriched uranium-plutonium oxide powder from Westinghouse	Building 235-F
Enriched uranium-plutonium oxides (high-fired) from Oak Ridge	FB-Line
Enriched uranium-plutonium oxides (high-fired) from Hanford	Building 235-F
Enriched uranium-plutonium oxides from Hanford	FB-Line
Enriched uranium-plutonium oxides, pellets, powder from Hanford	Building 235-F
Enriched uranium-plutonium-natural uranium oxide from Oak Ridge	Building 235-F
Enriched uranium-plutonium-neptunium compounds from Livermore	FB-Line
Plutonium-enriched uranium (passivated) alloy from Argonne	Building 235-F
Plutonium-enriched uranium alloy from Argonne	FB-Line
Plutonium-enriched uranium oxide from Los Alamos	Building 235-F
Plutonium-enriched uranium oxides from Rocky Flats	Building 235-F
Plutonium-neptunium compounds from Livermore	FB-Line
Plutonium-neptunium oxide from Hanford	FB-Line
Scrap and residue mixed solids - Approximately 200 packages containing less than 100 grams (3.5 ounces) of plutonium or neptunium per package:	
Enriched uranium and plutonium oxides from Battelle	Building 235-F
Enriched uranium and plutonium oxides from Hanford	Building 235-F
Enriched uranium-neptunium-aluminum scrap (desicooler packaging)	Building 235-F
Enriched uranium-plutonium alloy from Argonne	FB-Line
Enriched uranium-plutonium and natural uranium-plutonium oxides from Battelle	Building 235-F
Enriched uranium-plutonium compound from Argonne	Building 235-F
Enriched uranium-plutonium compounds from Battelle	235-F, FB-Line
Enriched uranium-plutonium compounds from Los Alamos	Building 235-F
Enriched uranium-plutonium from Argonne	Building 235-F
Enriched uranium-plutonium oxides from Hanford	FB-Line
Enriched uranium-plutonium reject fuel rods from Vallecitos	Building 235-F
Enriched uranium-plutonium-thorium alloy with zirconium cladding	Building 235-F
Enriched uranium-plutonium-titanium alloy (passivated) and glass from Argonne	Building 235-F
Enriched uranium-plutonium-titanium in zirconium oxide crucible from Argonne	Building 235-F
Enriched uranium-plutonium-zirconium alloy from Argonne	Building 235-F
Enriched uranium-plutonium-zirconium compound from Argonne	Building 235-F
Enriched uranium-plutonium-zirconium oxides from University of Virginia	Building 235-F

TC

Table A-1. (continued).

Description and storage management activities	Location
Enriched uranium-zirconium alloy from Argonne	Building 235-F
Plutonium-enriched uranium compound from Nuclear Energy	FB-Line
Plutonium-enriched uranium compound from Oak Ridge	FB-Line
Plutonium-enriched uranium-thorium alloy from Argonne	Building 235-F
Plutonium-neptunium-curium-americium compounds	FB-Line
Plutonium-thorium alloy from Battelle	Building 235-F
Plutonium-thorium compounds from Battelle	Building 235-F
Plutonium-thorium compounds from Hanford	FB-Line
Scrap (high-fired enriched uranium oxide) from Hanford	FB-Line
Plutonium-238 scrap materials - Approximately 120 packages of material containing small quantities of plutonium-238, mostly in the form of plutonium oxide.	
Plutonium-238 miscellaneous solids and nickel-coated oxide spheres from Mound and Rocky Flats	235-F
Plutonium-238 scrap materials from H-Area	HB-Line Vaults
Plutonium-238 scrap material containing iron oxide	Old HB-Line
Plutonium-238 oxide and compounds from program uses of plutonium-238	SRTC

	Fuel/target in inventory		Storage activities	
Location	Number	Units	Number of storage positions	Storage units
TC **Mark-31 targets** - Approximately 16,000 target slugs, containing 147 metric tons (160 tons) of nuclear material (primarily uranium-238 and plutonium-239) clad with aluminum. Most targets are in reactor basins in stainless-steel buckets within stainless-steel boxes equipped with a loose-fitting lid.				
F-Canyon Basin	2,448	Slugs	34	Buckets
K-Reactor Disassembly Basin	180	Slugs	3	Buckets
L-Reactor Disassembly Basin (Continuation box not irradiated)	12,496	Slugs	174	Buckets in SS box
L-Reactor Disassembly Basin (Continuation box not irradiated)	720	Slugs	10	Buckets in SS box
Receiving Basin for Offsite Fuel	40	Slugs	40	Slugs
Taiwan Research Reactor fuel - Approximately 81 cans of leaking TRR fuel, containing 20 metric tons (22 tons) of natural uranium metal with plutonium-239 clad with aluminum, are stored in RBOF.				
Receiving Basin for Offsite Fuel	81	Cans	81	Cans
Experimental Breeder Reactor II targets - One leaking can of EBR-II slugs, containing 17 metric tons (19 tons) of depleted uranium metal clad with aluminum, are stored in RBOF.				
Receiving Basin for Offsite Fuel	1	Can	1	Can
Enriched Uranium - Approximately 3,350 enriched uranium-aluminum alloy fuel tubes clad with aluminum.				
Mark-16 fuels				
K-Reactor Disassembly Basin	2	Tubes	2	Tubes
L-Reactor Disassembly Basin	516	Assemblies	137 146	Bundles Assemblies

Table A-1. (continued).

Location	Fuel/target in inventory		Storage activities	
	Number	Units	Number of storage positions	Storage units
P-Reactor Disassembly Basin	53	Assemblies	7	Bundles
H-Canyon	13	Assemblies	13	Assemblies
Mark-22 fuels				
K-Reactor Disassembly Basin	900	Assemblies	62 432	Bundles Assemblies
P-Reactor Disassembly Basin	396	Assemblies	52	Bundles
Other aluminum-clad targets - About 1,800 aluminum-clad targets containing thorium to produce uranium-233, cobalt used as part of the reactor power control because it is a neutron absorber, thulium, monitor pins and slugs.				
Cobalt slugs				
K-Reactor Disassembly Basin	168	Slugs	168	Slugs
L-Reactor Disassembly Basin	602	Slugs	602	Slugs
P-Reactor Disassembly Basin	1	Slug	1	Slug
Mark-50A thorium elements containing uranium-233				
K-Reactor Disassembly Basin	16	Slugs	16	Slugs
L-Reactor Disassembly Basin	1	Slug	1	Slug
Irradiated thulium slugs				
L-Reactor Disassembly Basin	1	Slug	1	Slug
Irradiated aluminum-clad slugs (M-51) in quatrefoils				
P-Reactor Disassembly Basin	60	Slugs	60	Slugs
Mark-42 target assemblies				
P-Reactor Disassembly Basin	2	Assemblies	2	Assemblies
Plutonium-242 flux monitor pins				
P-Reactor Disassembly Basin	65	Slugs	65	Slugs

TC

Lithium-aluminum reactor materials - Approximately 420 unirradiated control rods, spargers, and targets and irradiated control rods stored in reactor disassembly basins. Construction materials are lithium-aluminum alloy clad with aluminum.

Location	Storage activities
Lithium-aluminum control rods, spargers, and targets[b]	K-, L-, P-Reactor Disassembly Basins

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a. Abbreviations: Al = aluminum; Am = americium; Be = beryllium; Ca = calcium; DU = depleted uranium; EBR = Experimental Breeder Reactor; HEU = highly enriched uranium; LEU = low enriched uranium; LWR = light-water reactor; MEU = moderately enriched uranium; NU = natural uranium; Pu = plutonium; RBOF = Receiving Basin for Offsite Fuel; SRTC = Savannah River Technology Center; SS = stainless steel; Th = thorium; TRR = Taiwan Research Reactor; U = uranium; Zr = zirconium.

b. The irradiated lithium-aluminum materials contain tritium produced in the irradiation process. These materials will be processed by the SRS Tritium Facilities. Tritium and associated facilities are not included in the scope of this EIS (see Chapter 1). The unirradiated lithium-aluminum materials do not contain radioactive material. DOE will store these materials until it can make disposition decisions. This table includes all of the lithium-aluminum materials in the reactor disassembly for completeness.

A.3 Quantities of SRS Materials

A.3.1 QUANTITIES OF WEAPONS-USABLE MATERIALS

The Secretary of Energy recently released the following information (DOE 1994i) on the historic production and current inventories of plutonium. The United States produced 89 metric tons (98 tons) of weapon-grade plutonium from 1945 to 1988. The Savannah River Site produced 36 metric tons (40 tons) of this total between 1953 and 1988. At present, the SRS has 2.1 metric tons (2.3 tons) of plutonium in inventory, which includes the following:

- 0.5 metric ton (0.6 ton) in the form of metals, sources, and standards
- 1.3 metric tons (1.4 tons) in the form of residues
- 0.3 metric ton (0.33 ton) contained in irradiated materials

None of the activities proposed by DOE in this EIS would result in the production of additional amounts of plutonium. All of the plutonium-bearing materials at the SRS that are in the scope of this EIS have been included in the 2.1-metric-ton (2.3-ton) inventory. Actions proposed by DOE in this EIS would only change the relative distribution listed above.

L11-8

Similarly, the Secretary of Energy released the following information on the historic production and current inventories of highly enriched uranium. The United States produced 994 metric tons (1,096 tons) of highly enriched uranium from 1945 to 1992. The current DOE inventory is approximately 260 metric tons (287 tons). The SRS has 24.4 metric tons (26.9 tons) stored as metal, irradiated fuel, unirradiated fuel, oxide, and other forms (DOE 1994i).

None of the activities proposed by DOE in this EIS would result in the production of additional amounts of highly enriched uranium. All of the materials at the SRS that are in the scope of this EIS and that contain highly enriched uranium have been included in the 24.4-metric-ton (96.9-ton) inventory. Some of the actions proposed by DOE in this EIS would change the physical form of the highly enriched uranium. For example, processing irradiated fuel to separate the highly enriched uranium from fission products, converting the highly enriched uranium to an oxide, and packaging it into drums for storage. DOE's preferred alternatives for fuels and existing solutions containing highly enriched uranium would result in diluting the material with depleted uranium to produce an oxide of low enriched uranium. This would reduce the 24.4-metric-ton (96.9-ton) inventory.

DOE cannot provide specific information on the amount of plutonium or highly enriched uranium stored in individual containers or locations because such information is sensitive in relation to the protection of the material against theft, diversion, or acts of sabotage.

A.3.2 QUANTITIES OF IRRADIATED SPENT FUEL AND TARGETS L11-8

The SRS has a large number of irradiated spent fuel and targets stored in the K-, L-, and P-Reactor disassembly basins in the wet storage areas of the canyons, and in the Receiving Basin for Offsite Fuel. The spent fuel and targets at the SRS represent approximately 200 metric tons of heavy metal (MTHM) (200 tons).

APPENDIX B. RADIOACTIVITY AND RADIOLOGICAL HEALTH EFFECTS

Since the discovery of radioactivity in the late 1800s, scientists have studied this natural phenomenon extensively. Its potential for commercial and medical benefits – and its health risks – quickly became apparent. In comparison with many nonradioactive chemicals, radioactivity is easy to detect and measure. Hundreds of studies have quantified its effects on living organisms.

B.1 WHAT IS RADIOACTIVE MATERIAL?

Material is "most comfortable" or stable when it is in a form that is originally found in nature. For example, when iron ore (iron oxide) is removed from the earth and refined into steel, the steel is really an "unnatural" form of iron. If we throw a piece of shiny refined steel into a dump, the iron attempts to return to its normal form by rusting. (Rusted iron is iron oxide, the same form that the iron was in before being refined.) Materials whose atoms are in an excited state are frequently radioactive. A radioactive atom attempts to become more nearly "normal" by emitting energy or small particles from its nucleus. Material that contains excited atoms that are radioactive is known as radioactive material. Some radioactive material occurs naturally. This "naturally radioactive" material was raised to an excited state millions of years ago when the Earth was formed. Other materials are made radioactive when they are placed in a nuclear reactor or are bombarded with nuclear particles from an accelerator. All of these radioactive materials, natural or manmade, emit energy or particles from their nuclei to become more nearly stable or normal. L11-6 L11-14 L11-16

B.2 WHAT IS RADIATION?

Radiation is the energy or particles emitted by radioactive material in its attempt to become more stable. Although the generic term "radiation" includes microwaves, radio waves, and visible light, we use this term to refer to the high energy or particle form emitted from radioactive material that is more accurately called "ionizing radiation." Ionizing radiation is called that because it is strong enough to break molecules into smaller pieces called ions. Ionizing radiation produces energy that can be useful but that also can damage living tissue.

B.3 KINDS OF RADIATION

There are four major types of radiation:

- **Alpha particles** are heavy particles, consisting of two neutrons and two protons. Because the particles are slow moving as well as heavy, alpha radiation can be blocked by a sheet of paper. However, once an alpha emitter is in living tissue, it can cause substantial damage.

- **Beta particles** consist of single electrons. They are moderately penetrating and can cause skin burns from external exposure to high levels; they can be blocked by a sheet of plywood.

- **Gamma rays** are high-energy electromagnetic rays similar to X-rays. They are highly penetrating. Several inches of lead or several feet of concrete are required to shield against gamma rays. Gamma rays can be very damaging to living tissue at high dose rates.

- **Neutrons** are particles that can be both penetrating and very damaging to living tissue, depending on their energy and dose rate.

L11-6
L11-14
L11-16

B.4 MEASURING RADIOACTIVITY

One way to measure radioactive material is to measure the rate at which the radioactive atoms disintegrate. This mechanical measurement uses the "curie" as the basic unit, 1 curie being 37 billion atomic disintegrations in 1 second.

Another way to measure the impact of radioactive material on humans is to calculate the energy deposited by the radiation emitted from radioactive material when the radiation interacts with human tissue. This is a health-based approach. Its basic unit of measurement is the rem. This approach is based on actual damage observed from exposure to very high levels of radiation or the accumulation of significant amounts of radioactive material in body organs such as bones or lungs. The calculation of energy deposited in tissue has some uncertainties, but this approach allows more meaningful measurements than measuring the disintegration rate of the radioactive material. Because a single radiation dose has different effects on different body organs, it is not easy to predict what effect a given dose will have on a person's health. This is particularly true when the doses are at levels well below those at which physical effects have been observed. The precise effects of low-level radiation exposure continue to be controversial.

B.5 HALF-LIFE

The more highly excited a radioactive atom is, the more rapidly it breaks down or decays, and the shorter its half-life. The half-life is the time required for the radioactivity of a quantity of a specific radioactive nuclide to decay to one half its original activity. During a second half-life, the radioactivity will decay to one-half of that value, or one-quarter of the starting activity, and so on. The half-lives of radionuclides range from fractions of a second to billions of years. Radionuclides with short half-lives decay to negligible activities in short periods of time, but tend to emit higher energy radiation during decay, and thus produce a higher dose. Radionuclides with long half-lives, although present for much longer periods of time, tend to decay by emitting lower energy radiation.

B.6 PROPERTIES OF SRS RADIOACTIVE MATERIAL

Table B-1 summarizes the half-lives, types of radiation emitted, comparison of relative external and internal hazards, and the possibility of criticality for the various radioactive materials covered in this environmental impact statement.

B.7 HOW CAN RADIATION CAUSE DAMAGE?

L11-6
L11-14
L11-16

In living organisms, the chemical changes induced by high doses of radiation can lead to serious illness or death. At lower doses, radiation can produce chemical changes in cells that sometimes lead to cancerous growth of the cells, and can damage cellular DNA, sometimes leading to genetic mutations. Even the "natural background" radiation level, which depends on geographic location, altitude, and other factors, imposes some risk of illness. An estimated 82 percent of the average radiation exposure received by people in the United States comes from natural sources.

B.8 UNDERSTANDING RADIATION HAZARDS

Measuring a material's radioactivity or the dose its radiation produces in tissue is only the first step toward understanding its potential hazards to living organisms. Other important factors include the following:

- **Type of radiation.** Some radiation, such as alpha particles, can cause chemical changes at short range if they are inside the body. If alpha-emitting radioactive material is outside the body, it has little if any detrimental effect because the alpha radiation cannot penetrate the thin

Table B-1. Radiological properties of SRS radioactive material.

L11-6
L11-14
L11-16

Material	Half-life	Major radiation emitted during decay	Relative external hazard[a] (rem/hr per Ci @ 1m)	Relative internal hazard[b] (rem/Ci)	Is criticality a major concern?[c]
Americium-241	432 years	Alpha Gamma	0.31	520	No
Americium-243	7,380 years	Alpha Gamma	0.31	520	No
Cesium-137	30 years	Beta Gamma	0.38	0.032	No
Co-60	5.3 years	Gamma Beta	1.4	0.15	No
Curium-244	18 years	Alpha	0.064	270	No
Neptunium-237	2.1 million years	Alpha Beta Gamma	0.46	490	No
Plutonium-238	88 years	Alpha	0.079	460	No
Plutonium-239	24,000 years	Alpha	0.030	510	Yes
Plutonium-240	6,569 years	Alpha	0.075	510	No
Plutonium-242	376,000 years	Alpha	0.062	480	No
Thorium-230	77,000 years	Alpha Gamma	0.069	320	No
Thorium-232	14 billion years	Alpha Gamma	0.068	1,600	No
Uranium-233	159,000 years	Alpha	0.029	130	Yes
Uranium-234	245,000 years	Alpha	0.078	130	No
Enriched Uranium (uranium-235)	704 million years	Alpha Beta Gamma	0.34	120	Yes
Depleted Uranium (uranium-238)	4.5 billion years	Alpha Beta	0.065	120	No
Tritium	12.3 years	Beta	0	0.000063	No

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a. The relative hazard from external radiation is measured in units of rem per hour of gamma radiation at 1 meter from 1 curie of each radionuclide. (Data were taken from *The Health Physics and Radiological Health Handbook* (Scinta 1992). Values from this reference were multiplied by 3,700 to convert them from milliSieverts per 1,000 Becqueral at 1 meter to rem per hour per curie at 1 meter.)

b. The relative hazard from inhalation of each radionuclide is measured as the 50-year Committed Effective Dose Equivalent in units of rem per microcurie of intake. (Data are dose conversion factors from DOE 1988.)

c. The concern for criticality in this case is based on the material's ability to undergo fission following absorption of a thermal neutron. DOE recognizes that some of the listed radionuclides can undergo fission following absorption of fast neutrons; however, DOE does not anticipate that conditions would be such that accidental criticality of these materials would be a major concern. Therefore, DOE has identified only those radionuclides that can undergo thermal fission as a major criticality concern.

outermost layer of dead skin. Other kinds of radiation, such as gamma rays, can be harmful from distant external sources.

- **Energy of the radiation.** The energy associated with radiation can vary from relatively low to relatively high levels. In general, the higher the energy level, the higher the dose produced when the radiation interacts with tissue.

- **Chemical stability.** Radioactive substances that can burn or otherwise react are more susceptible to being dispersed in the environment. For instance, some forms of plutonium can ignite spontaneously if exposed to air.

- **Biological uptake.** Radioactive elements incorporated into organisms are more harmful than those that pass through the organism quickly. Many radioactive elements are readily absorbed into bone or other tissues. Radioactive iodine is concentrated in the thyroid, while radium and strontium are deposited in the bone. Insoluble particles like plutonium oxide can remain in lung tissue for very long periods of time. L11-6 L11-14 L11-16

- **Dose and dose rate.** Dose rate is the amount of radiation received in a given time period, such as rem per day. In general, the risks of adverse health effects are lower when exposure is spread over a long time period than when the same dose is received in a short time period.

- **Dose location.** Some kinds of living tissue (e.g., blood-forming tissue) are more sensitive to radiation than others (e.g., nerve tissue).

The combined effect of these factors makes the risk posed by even a simple radiation exposure difficult to estimate. Some radioactive materials contain many different radionuclides – along with various chemicals introducing even more uncertainty. In ICRP-60 (ICRP 1991), the International Commission on Radiological Protection recommended the use of lifetime risk factors for excess cancer deaths of 0.0005 excess cancer death per rem of exposure for the general population, and 0.0004 excess cancer death per rem of exposure for radiation workers. (The larger value for the general population occurs because that population includes children while the radiation worker population includes only people older than 18.)

APPENDIX C

FACILITY AND PROCESS DESCRIPTIONS

TABLE OF CONTENTS

Section Page

C.1 Fuel and Target Fabrication (M-Area) C-1

C.2 Reactors C-1

C.3 Chemical Separations (F-Canyon and H-Canyon) C-5

- C.3.1 F-Canyon (PUREX) Process C-10
- C.3.2 H-Canyon Process C-14

C.4 FB-Line C-17

C.5 HB-Line C-21

- C.5.1 Phase I C-23
- C.5.2 Phase II C-23
- C.5.3 Phase III C-23

C.6 Outside Facilities (F- and H-Areas) C-24

- C.6.1 Chemical Storage C-24
- C.6.2 Water Handling Facilities C-24
- C.6.3 Acid Recovery Unit C-26
- C.6.4 General-Purpose Waste Tanks C-26
- C.6.5 General-Purpose Evaporator C-26
- C.6.6 Waste Handling Facilities C-27

C.7 A-Line (F-Area) C-27

C.8 A-Line (H-Area) C-28

C.9 Receiving Basin for Offsite Fuel C-28

C.10 Laboratories C-29

- C.10.1 Savannah River Technology Center C-30
- C.10.2 Other Laboratories C-30

C.11 Building 235-F C-30

C.12 Depleted Uranium Storage Facilities C-31

C.13 Building 247-F Vault C-32

C.14 New Facilities C-32

- C.14.1 Uranium Solidification Facility C-32
- C.14.2 Dry Storage Facilities C-34
  - C.14.2.1 Spent Nuclear Fuel Storage Using a Dry Vault C-35
  - C.14.2.2 Spent Nuclear Fuel Storage Using Dry Casks C-36

TABLE OF CONTENTS (continued)

Section Page

C.14.3 Actinide Packaging and Storage Facility C-41
C.14.4 FB-Line Modification C-45
C.14.5 F-Canyon Vitrification Facility C-46
C.14.6 Low Enriched Uranium Oxide Waterhouse C-48

C.15 Waste Management Facilities C-49

C.15.1 Liquid High-Level Radioactive Waste C-56
C.15.2 Low-Level Solid Waste C-59
C.15.3 Transuranic Solid Waste C-61
C.15.4 Hazardous and Mixed Waste C-65
C.15.5 Low-Level Liquid Wastes C-67

TABLE OF CONTENTS (continued)

LIST OF FIGURES

Figure Page

C-1 Historic nuclear materials production cycle at the Savannah River Site C-2
C-2 M-Area C-3
C-3 Typical SRS Reactor - K-Reactor and surrounding area C-4
C-4 Rail car with cask C-6
C-5 F-Canyon and surrounding area C-7
C-6 H-Canyon and surrounding area C-8
C-7 F-Canyon building sections C-9
C-8 Standard canyon process vessel C-11
C-9 Historic PUREX process flow C-12
C-10 Historic H-Canyon process flow C-15
C-11 F-Canyon and FB-Line facilities C-18
C-12 FB-Line worker using a glovebox C-19
C-13 FB-Line process flow C-20
C-14 H-Canyon and HB-Line facilities C-22
C-15 F-Canyon, showing Outside Facilities C-25
C-16 Layout of a modular vault dry storage facility for spent nuclear fuel C-37
C-17 Layout of a modular dry cask storage facility for spent nuclear fuel C-40
C-18 Proposed location of Actinide Packaging and Storage Facility C-43
C-19 Savannah River Site, showing waste management facilities C-50
C-20 F-Area tank farm C-51
C-21 H-Area tank farm C-52
C-22 Liquid high-level waste handling C-53
C-23 Low-Level Radioactive Waste Disposal Facility (E-Area) C-54
C-24 Transuranic waste storage pads (E-Area) C-55
C-25 Hazardous waste storage facility (B-Area) C-57
C-26 Mixed waste storage building (E-Area) C-58
C-27 SRS waste handling processes C-59
C-28 F/H-Area Effluent Treatment Facility (H-Area) C-60
C-29 Long-lived waste storage building (E-Area) C-62
C-30 Low-activity waste vault (E-Area) C-63
C-31 Intermediate-level nontritium and tritium waste vaults (E-Area) C-64
C-32 Transuranic waste storage (E-Area) C-66

APPENDIX C. FACILITY AND PROCESS DESCRIPTIONS

This appendix describes the principal facilities associated with the nuclear materials described in this environmental impact statement. The operations described are historic; the descriptions do not necessarily reflect how DOE would implement the alternatives discussed in this EIS. Figure C-1 shows the historic cycle and facilities used to produce, process, and store nuclear materials at the Savannah River Site. Chapter 2 describes the operations that would be associated with the alternatives, and includes short descriptions of proposed facilities or major modifications of SRS structures that would affect the alternatives, and of waste management facilities that would process wastes associated with stabilizing nuclear materials. TC

C.1 Fuel and Target Fabrication (M-Area)

M-Area (see Figure C-2) contains facilities used historically to fabricate fuel, special targets, and components for SRS production reactors. The facilities contain conventional equipment for melting, casting, and shaping metal, including furnaces, extrusion presses, lathes, handling equipment, and storage racks.

Buildings 313-M, 321-M, and 320-M contain the equipment used to fabricate depleted uranium targets, reactor fuel, and tritium targets, respectively. Building 321-M also contains the extrusion presses and finishing equipment that DOE used to extrude neptunium-237 oxide billets into neptunium targets, which were irradiated to produce plutonium-238. "Deinventory" of the facility (i.e., packaging unused nuclear materials and placing them in storage at the SRS or returning them to their sources) is underway. Buildings 313-M, 320-M, and 322-M (the Metallurgical Laboratory) have been deinventoried. Building 321-M is being deinventoried at present.

The SRS received raw aluminum, uranium, lithium, etc., at Building 315-M from commercial vendors and other DOE sites. The raw materials were cast, extruded, and machined into long cylindrical tubes or short cylindrical slugs of metal, depending on whether the reactor component was fuel or target. After fabrication, the fuel and targets were shipped to a reactor area (C, K, L, P, or R) for irradiation.

C.2 Reactors

Of the five production reactors constructed at the SRS in the early 1950s, four (C, L, P, and R) have been permanently shut down, and one (K-Reactor) is in indefinite "cold standby" (Figure C-3). R-Reactor is scheduled for decontamination and decommissioning. TE

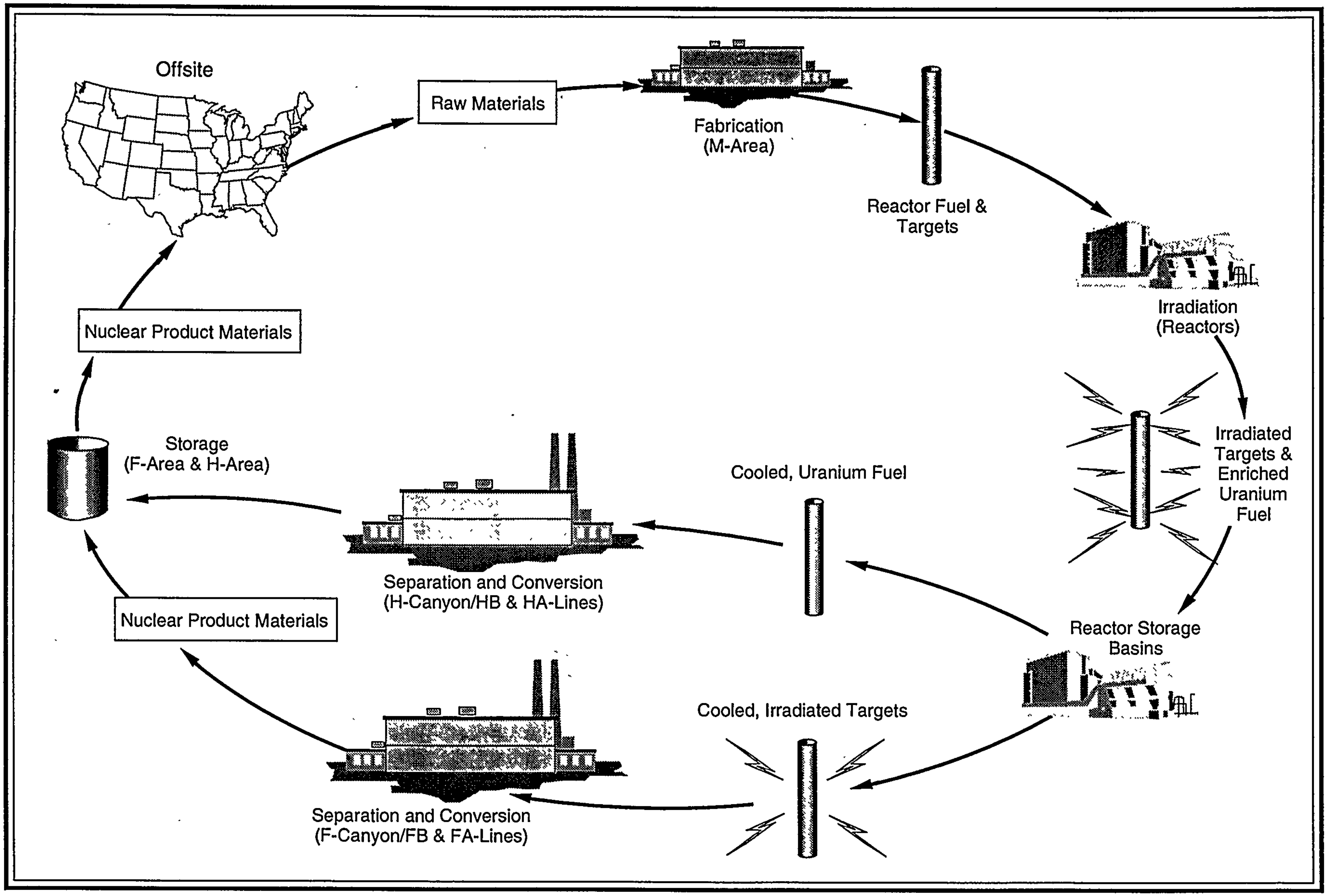


Figure C-1. Historic nuclear materials production cycle at the Savannah River Site.

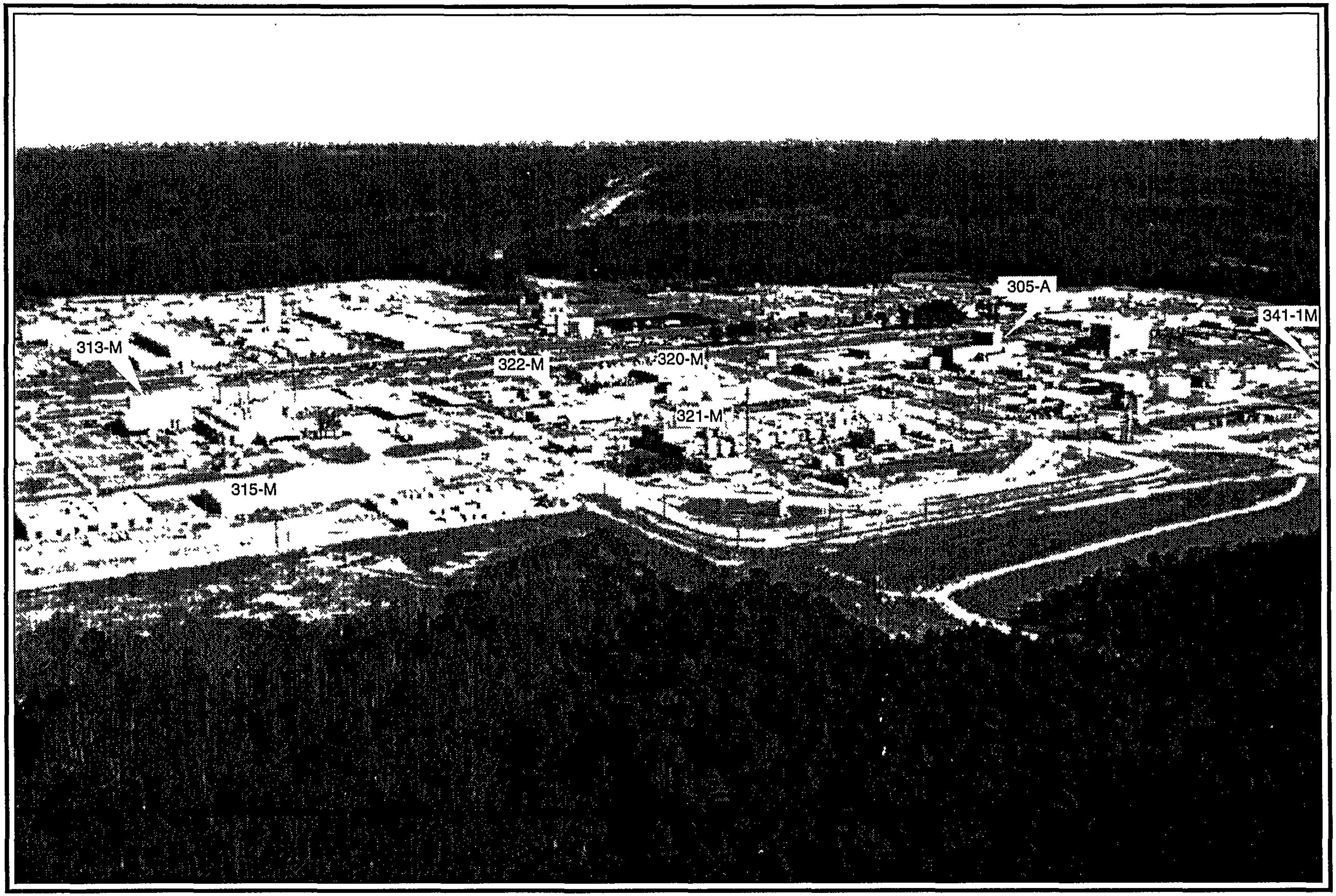


Figure C-2. M-Area (view toward the southeast).

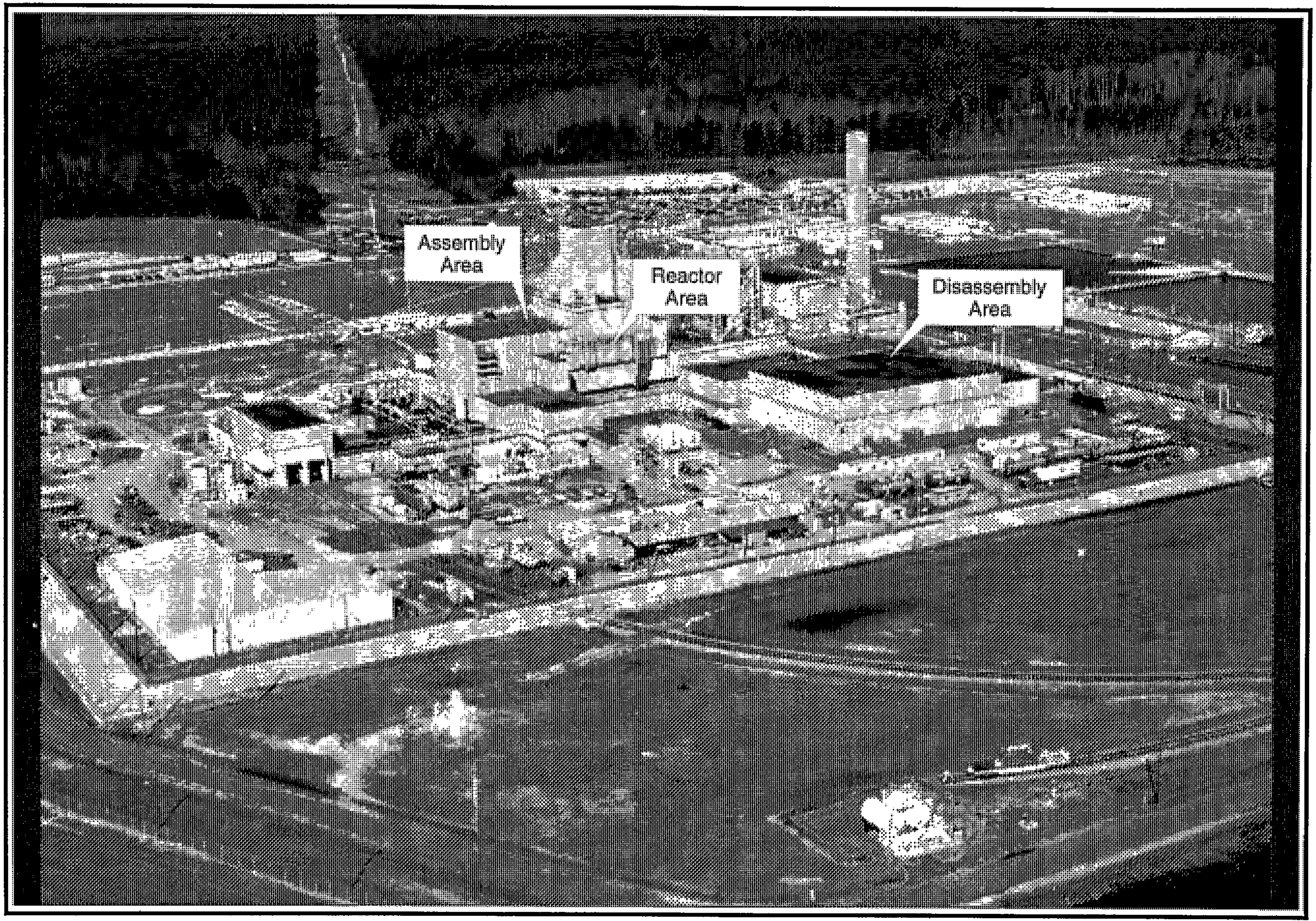


Figure C-3. Typical SRS reactor - K-Reactor and surrounding area (view toward the northeast).

Each reactor has an assembly area for the receipt, handling, and storage of new (i.e., unirradiated) fuel and targets. Racks and vaults store new fuel and targets. Similarly, each reactor has a disassembly area for the storage, handling, and shipment of irradiated fuel and targets that have been removed from the reactor. The disassembly area consists primarily of water-filled basins with metal racks designed for vertical or horizontal storage of fuel tubes, and metal buckets for storing targets. The disassembly basins are about 49 meters (160 feet) wide, 67 meters (220 feet) long and 5 to 9 meters (17 to 30 feet) deep. The volume of water in the basins ranges from 12,800,000 to 18,200,000 liters (3,380,000 to 4,800,000 gallons). The K- and L-Reactor disassembly basins are identical; the P-Reactor basin is the largest. The basins are constructed of unlined concrete coated with vinyl paint. Each has systems for circulating, filtering, and deionizing the water to maintain proper chemistry. Cranes, rigging, and handling equipment in the disassembly area can move or load fuel in casks for shipment to other areas on the Site.

Fuel and targets from M-Area were placed in storage racks or concrete vaults, then were grouped into assemblies and placed in a reactor core. The irradiation of the targets and fuel produced special isotopes. The irradiation time depended on the isotope to be produced. After their removal from the reactor core, the targets and fuel were placed in the water-filled basin to cool the fuel and targets and to allow the decay of short-lived radioactive products. The water also provided radiation shielding to operating personnel. After the targets or fuel had cooled for a brief period (12 to 18 months), they were disassembled and loaded in heavily shielded casks on rail cars (see Figure C-4), which were transferred to F- or H-Area for further processing.

C.3 Chemical Separations (F-Canyon and H-Canyon)

The similar F- and H-Canyon facilities use radiochemical processes for the separation and recovery of plutonium, neptunium, and uranium isotopes. The F-Canyon separated plutonium, irradiated natural or depleted uranium, and radioactive decay products. H-Canyon recovered uranium, highly enriched uranium-235, neptunium-237, and plutonium-238 from irradiated reactor fuels and targets. The following paragraphs apply to both canyons unless noted.

The F- and H-Canyons (see Figures C-5 and C-6; Figure C-6 also shows the Defense Waste Processing Facility in the adjoining S-Area) are reinforced concrete structures, 255 meters (836.6 feet) long, 37 meters (308 feet) wide, and 20 meters (121.4 feet) high. They are named for the two areas ("canyons") in each structure that house the large equipment (tanks, process vessels, evaporators, etc.) used in the chemical separations processes performed in each facility. The canyons are long (170 meters or 557.7 feet), narrow (an average of 6 meters or 19.7 feet), and deep

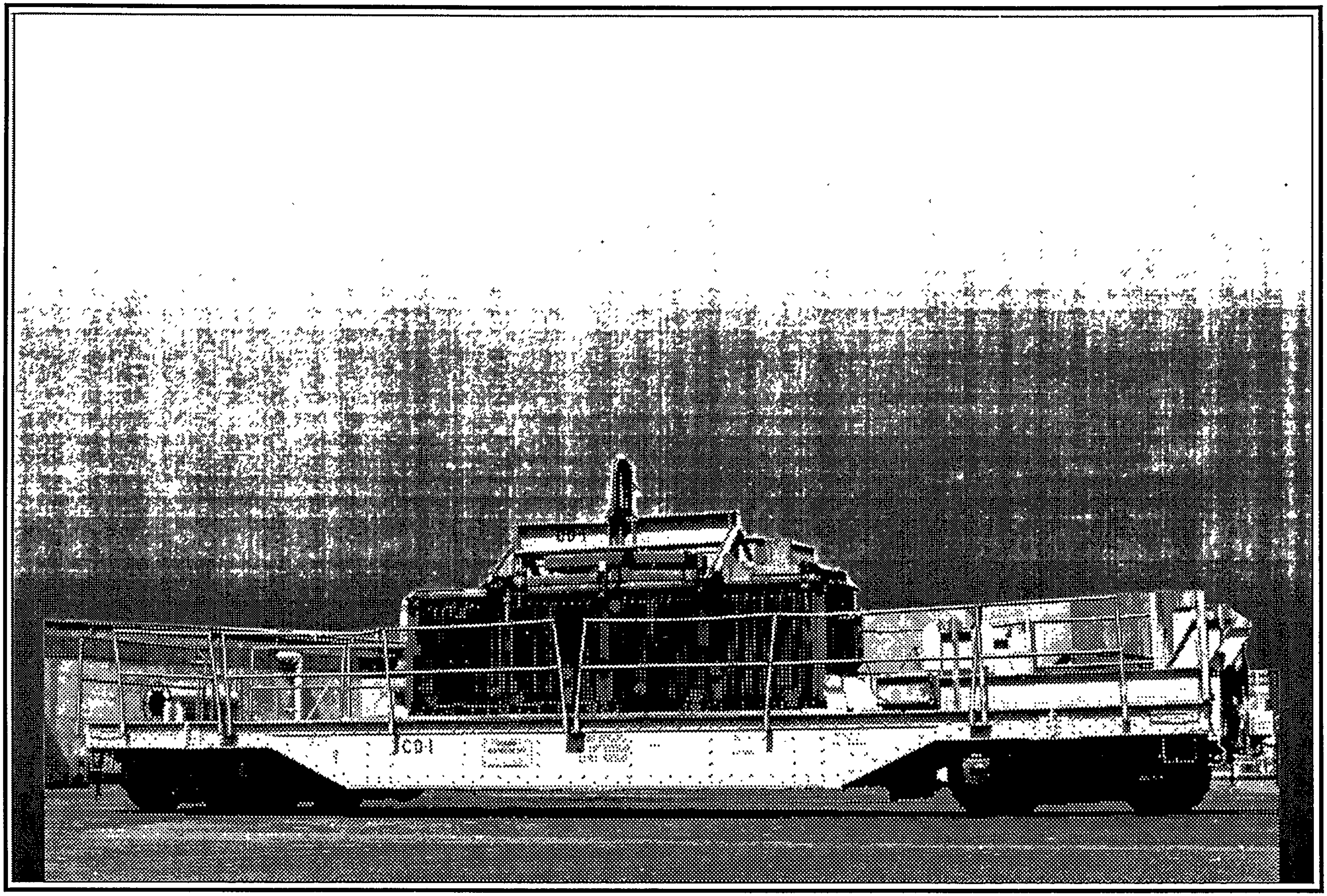


Figure C-4. Rail car with cask.

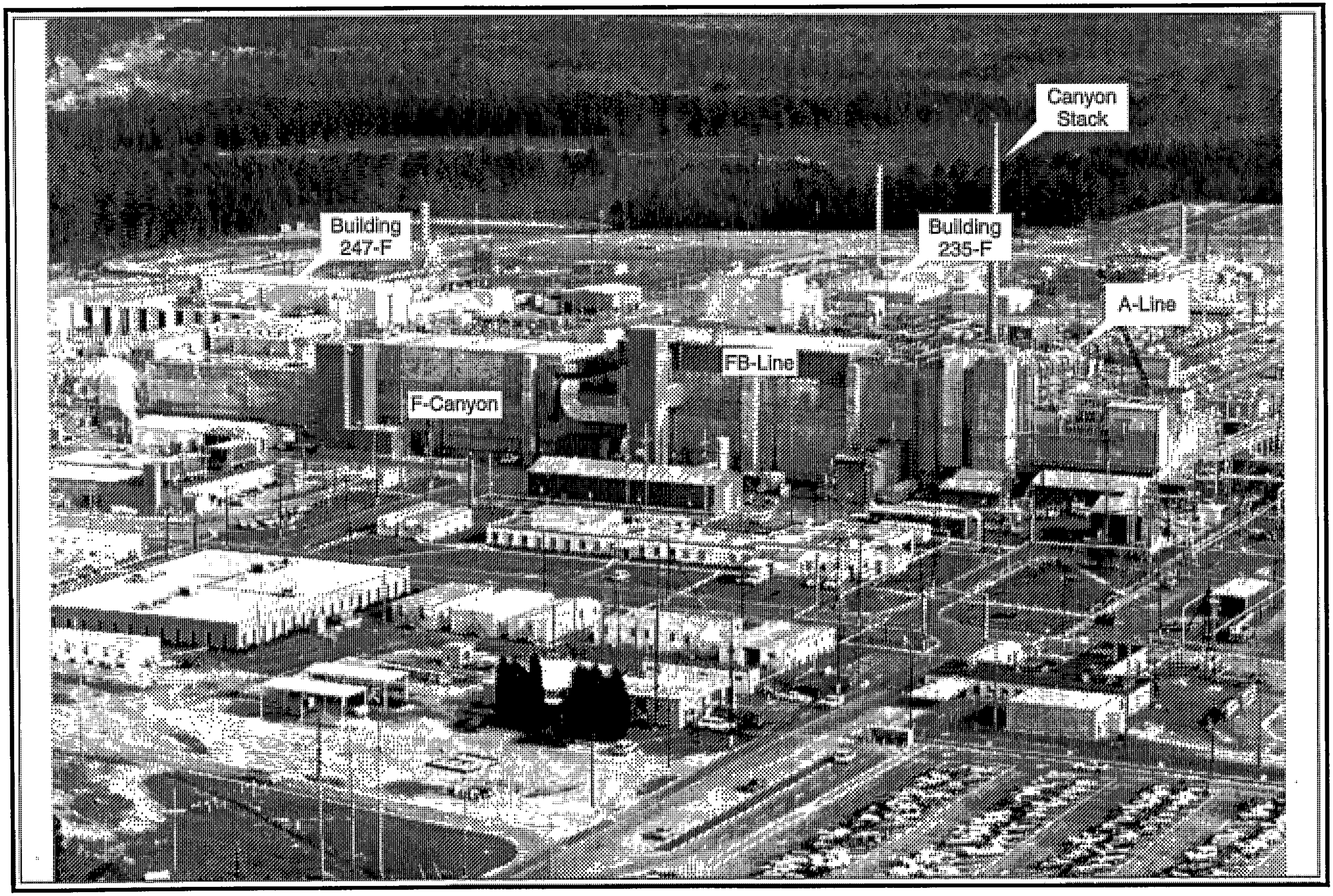


Figure C-5. F-Canyon and surrounding area (view toward northeast).

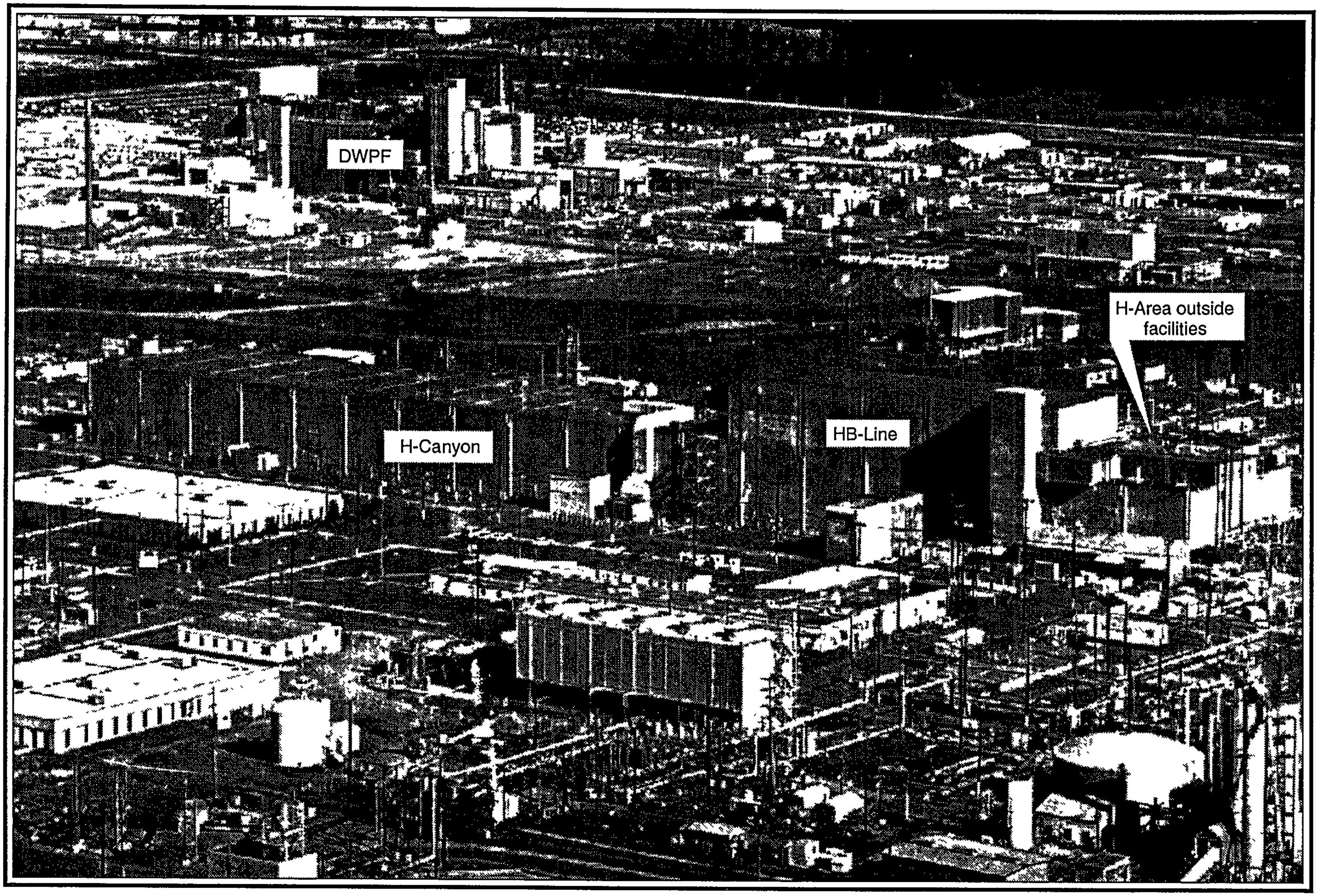


Figure C-6. H-Canyon and surrounding area (view toward northeast).

(20 meters or 65.6 feet). The "hot" and "warm" canyons in each facility are parallel and open from floor to roof. A center section, which has four floors or levels, separates the canyons. The center section contains office space, the control room for all facility operations, and support equipment such as ventilation fans. Figure C-7 is a cross-section view of a canyon facility. Processing operations involving high radiation levels (dissolution, fission product separation, and high-level radioactive waste evaporation) would occur in the hot canyon, which has thick concrete walls to shield people outside the facility and in the center section from radiation. The final steps of the chemical separations process, which generally involve lower radiation levels, would occur in the warm canyon.

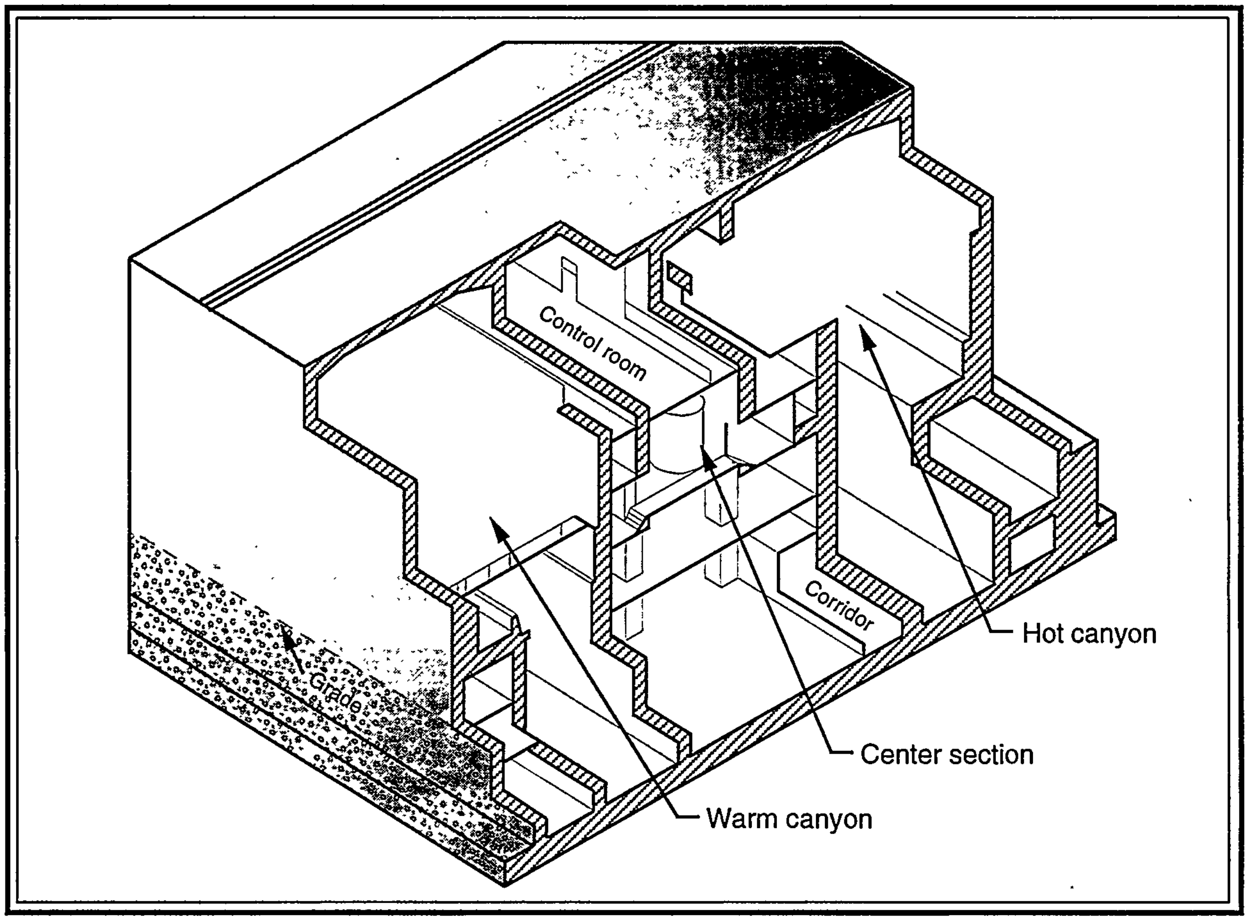


Figure C-7. F-Canyon building sections.

Services typical for a large industrial chemical facility are required to support F- and H-Canyon operations. For example, steam heats process vessels and is the motive force for transferring solutions through process cycles; lights, motors, control systems, etc., use electricity; compressed air provides pressure needed for various process monitoring systems (e.g., liquid level indicators) and powers some control systems; and a ventilation system provides conditioned air for the comfort of facility workers and for environmental control for the operation of sensitive equipment.

A separate ventilation system serves portions of the facility, such as the hot and warm canyons, that contain the radioactive process equipment. This system ensures the air pressure in such areas is below the pressure of the air outside the facility and the area occupied by workers. This design helps prevent the release of radioactive material outside the facility by ensuring that air always flows from outside to inside the process areas. Air in the process areas is exhausted from the facility through a large sand filter that removes 99.5 percent of any airborne radioactive material. A 61-meter (200-foot)-tall stack behind each canyon discharges this filtered air to the atmosphere and serves as the pathway for airborne emissions associated with the normal operation of the canyons.

There are two primary pathways for liquid effluents from the canyons:

- Condensates from secondary evaporators at the A-Line Outside Facilities containing low levels of radionuclides flow to the Effluent Treatment Facility (ETF) for further decontamination, if necessary, before their discharge to surface waters.
- A water system cools the hot and warm canyon process vessels. Underground pipes carry water to the canyons and distribute it. The water passes through coils inside the vessels (Figure C-8 shows a standard canyon process vessel) and flows back out of the canyon. Constant monitoring detects radioactivity in the water. If radioactivity is detected, the water is diverted to a treatment facility where the radioactivity is reduced below applicable limits before the water is discharged.

The equipment and processing stages in the canyons have been configured to separate and recover uranium and plutonium from irradiated fuel or targets, as described for each canyon in the following paragraphs.

C.3.1 F-CANYON (PUREX) PROCESS

The PUREX process consists of several major operations, referred to as "unit operations," which recover plutonium and uranium from irradiated reactor targets. The targets normally would be fabricated from uranium depleted in uranium-235 (e.g., at a level below the naturally occurring 0.711 weight percent). The irradiation process is designed to produce weapons-grade plutonium [i.e., plutonium that is greater than 93 percent plutonium-239, with the remainder of the plutonium isotopes similar to plutonium-240 and -241 (NAS 1994)]. The major unit operations are dissolution, head end, first cycle, second uranium cycle, and second plutonium cycle (see Figure C-9). Unit operations that support the product recovery operations are high-activity waste, low-activity waste, TE

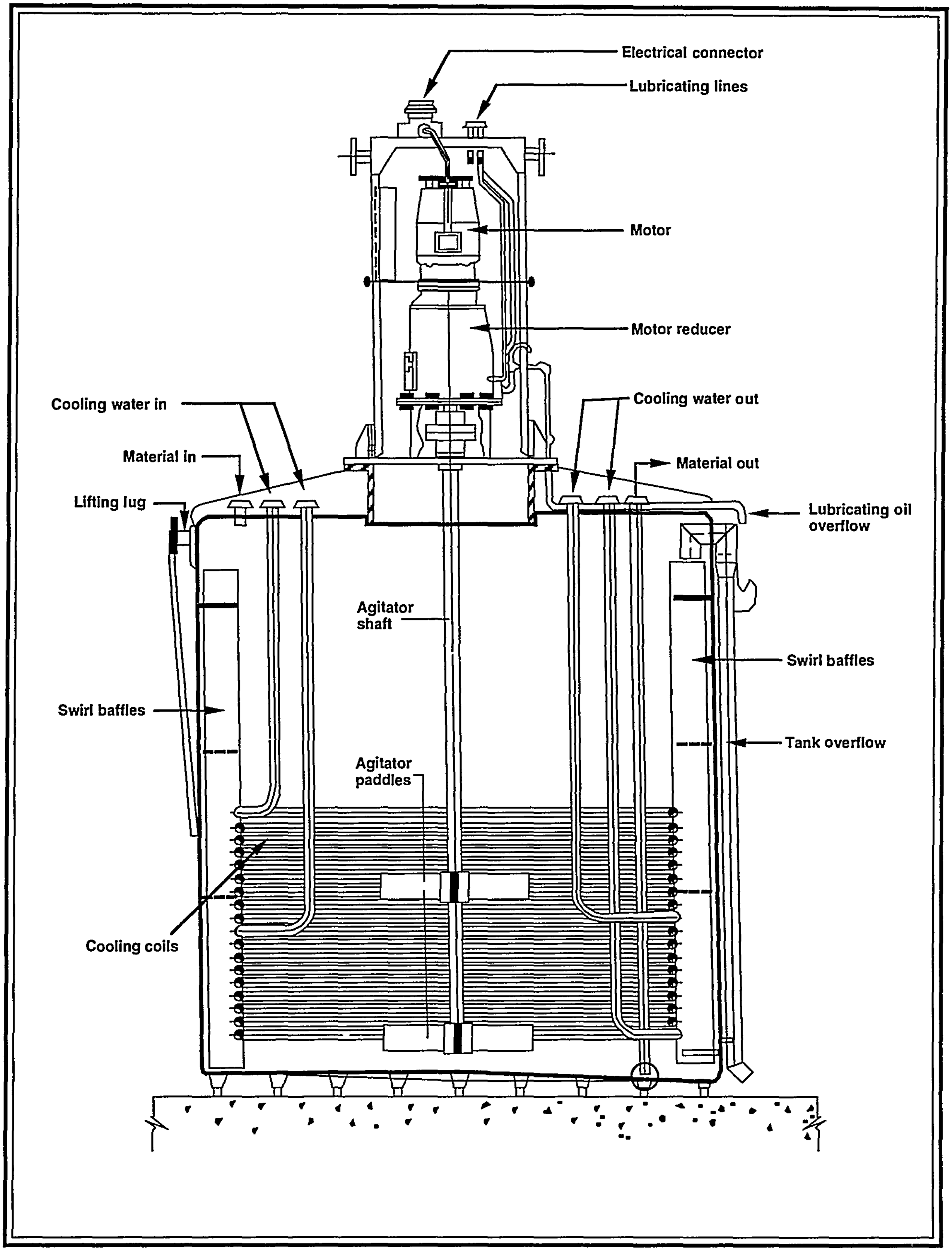


Figure C-8. Standard canyon process vessel.

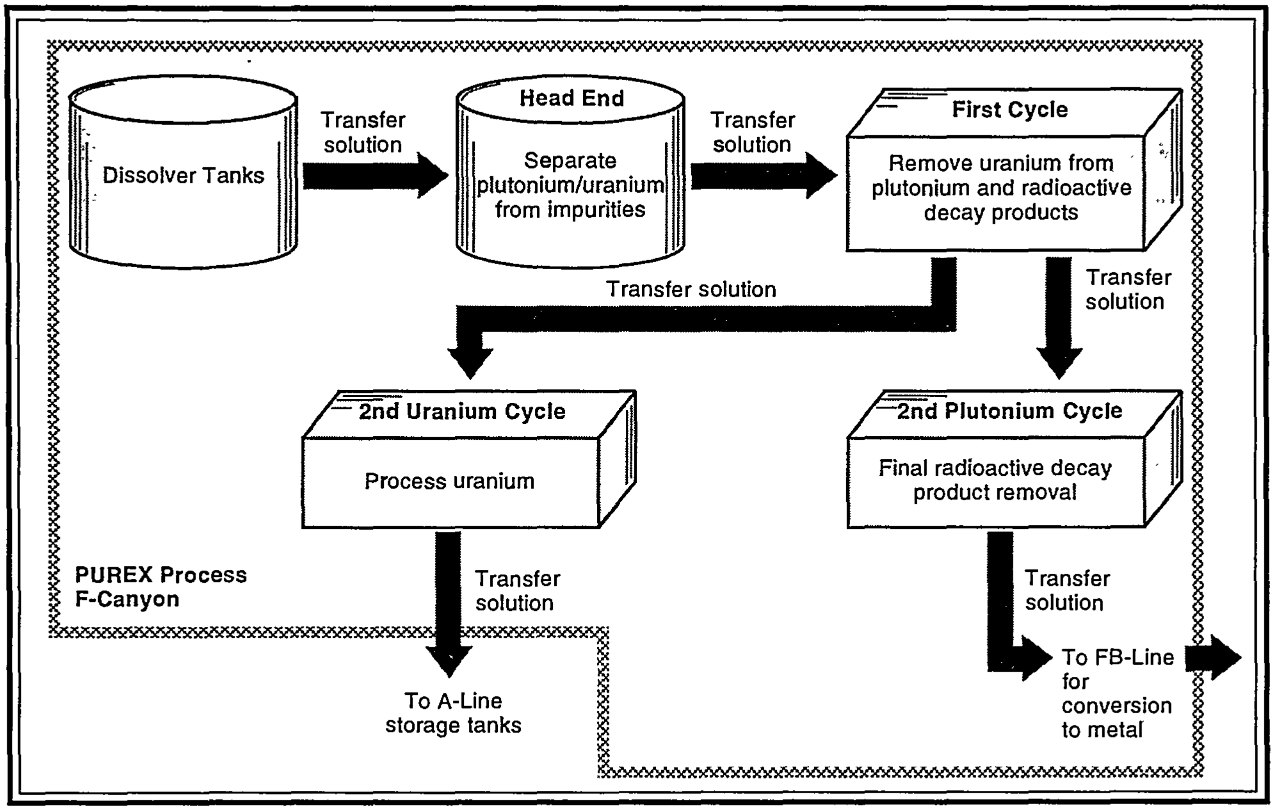


Figure C-9. Historic PUREX process flow.

solvent recovery, laboratory waste evaporation, etc. The F-Canyon process also has recovered neptunium-237 that results from PUREX process waste; this activity, which is no longer performed, is not part of this evaluation. Processes within the inner box are conducted in F-Canyon.

The following paragraphs describe major and support unit operations in F-Canyon:

- **Dissolution** - Irradiated targets on a rail car through an air lock are brought into the south end of the hot canyon. Each target consists of a cylinder of depleted uranium clad in aluminum. The targets have been irradiated in an SRS reactor to transform a portion of the depleted uranium into plutonium. Large water-filled casks on rail cars transfer the targets. The targets are removed from the casks and loaded into a large tank called a dissolver. Sodium hydroxide removes the aluminum cladding from the targets. The cladding solution is transferred to the high-level waste tanks. Heated nitric acid in the tank dissolves the target, resulting in a solution containing depleted uranium, plutonium, and radioactive decay products from the reactor irradiation process.

- **Head End** - This process occurs in two steps to prepare the target solution for uranium and plutonium separation. First, gelatin is added to precipitate silica and other impurities. Then the solution is transferred to a centrifuge where silica and other impurities are removed as waste. The clarified product solution is adjusted with nitric acid and water in preparation for the first cycle unit operation. The waste stream generated from the process is chemically neutralized and sent to the F-Area high-level waste tanks. The major components for this operation are a gelatin "strike" tank, a centrifuge feed tank, and a centrifuge.

- **First Cycle** - First cycle operation, which occurs in the hot canyon, has two functions: (1) to remove fission products and other chemical impurities, and (2) to separate the solution into two product streams (uranium and plutonium) for further processing. This separation process occurs as the product solution passes through a series of equipment consisting of a centrifugal contactor and mixer-settler banks. Before the introduction of the feed solution from the head end process, flows of solvent and acid solution are established in the equipment. When an equilibrium is established, the feed solution is introduced. The chemical properties of the acid/solvent/feed solutions in contact with each other cause radioactive decay products to separate from the uranium and plutonium. Later in the first cycle process, the plutonium is separated from the uranium in a similar manner. The first cycle produces four process streams: plutonium (with some residual radioactive decay products), which goes to the second plutonium cycle; a uranium solution (with some residual radioactive decay products), which goes to the second uranium cycle; a solvent stream, which goes to the solvent recovery cycle; and an aqueous acid stream, which goes to the high-level waste tanks. The acid stream contains most of the radioactive decay products. The equipment for this operation consists of a centrifugal contactor, mixer-settler banks, decanter tanks, and hold tanks.

- **Second Uranium Cycle** - The second uranium cycle (in the warm canyon) purifies the uranium solution from the first cycle and prepares the uranium for transfer to the FA-Line. The purification process is a separation process that occurs in a manner similar to that described for the first cycle. The uranium product solution, which contains a low concentration of radioactive decay products, is transferred from the warm canyon to storage tanks in the FA-Line facility, which is adjacent to the F-Canyon.

- **Second Plutonium Cycle** - The second plutonium cycle (in the warm canyon) purifies the plutonium solution from the first cycle by removing residual radioactive decay products, and prepares the plutonium for transfer to FB-Line. The purification process is a separation process that occurs in a manner similar to that described for the first cycle. The impurities are removed in an aqueous stream that goes to the low-activity waste unit operation for processing. The plutonium product solution, which contains a low concentration of radioactive decay products, is transferred to hold tanks for use as FB-Line feed material.

- **High- and Low-Activity Waste** - These unit operations reduce the volumes of the aqueous streams that contain radioactive decay products by using a series of evaporators in the hot and warm canyons. The feed to the evaporators originates with the primary separation process unit operations, such as the first cycle. The evaporator overheads, which contain most of the water and acid and very little of the radioactive decay product and chemicals used in solvent extraction, are transferred to tanks outside the building for acid recovery and recycling. The radioactive decay products and chemicals in the evaporator concentrate are neutralized and sent to the F-Area high-level waste tanks.

- **Solvent Recovery** - The primary purpose of this unit operation is to wash the solvent to remove impurities, and to recover the solvent and recycle it to solvent extraction cycles for reuse. This operation reconditions and removes impurities from the solvent. The impurities are transferred to low-activity waste for processing. A separate solvent recovery is used with each extraction cycle.

- **Laboratory Waste Evaporation** - The waste handling facilities receive high-level laboratory wastes from F-Area and the Savannah River Technology Center (SRTC) laboratories (see Section C.6.6) and transfer them to the warm canyon for evaporation. These wastes are evaporated and the recovered water is returned to the Outside Facilities for recycling and reuse. The concentrated waste is discharged to the F-Area high-level waste tanks.

C.3.2 H-CANYON PROCESS

The H-Canyon process consists of the recovery of highly enriched uranium (HEU) from reactor fuel and the recovery of neptunium-237 and plutonium-238 from targets. This EIS evaluates the highly enriched uranium, but not the neptunium-237 and plutonium-238 processing. The major unit operations associated with highly enriched uranium are dissolution, head end, first solvent extraction cycle, second uranium solvent extraction cycle, and second neptunium (or second actinide) solvent extraction cycle (see Figure C-10). Unit operations that support the product recovery operations are high-activity waste, low-activity waste, and solvent recovery.

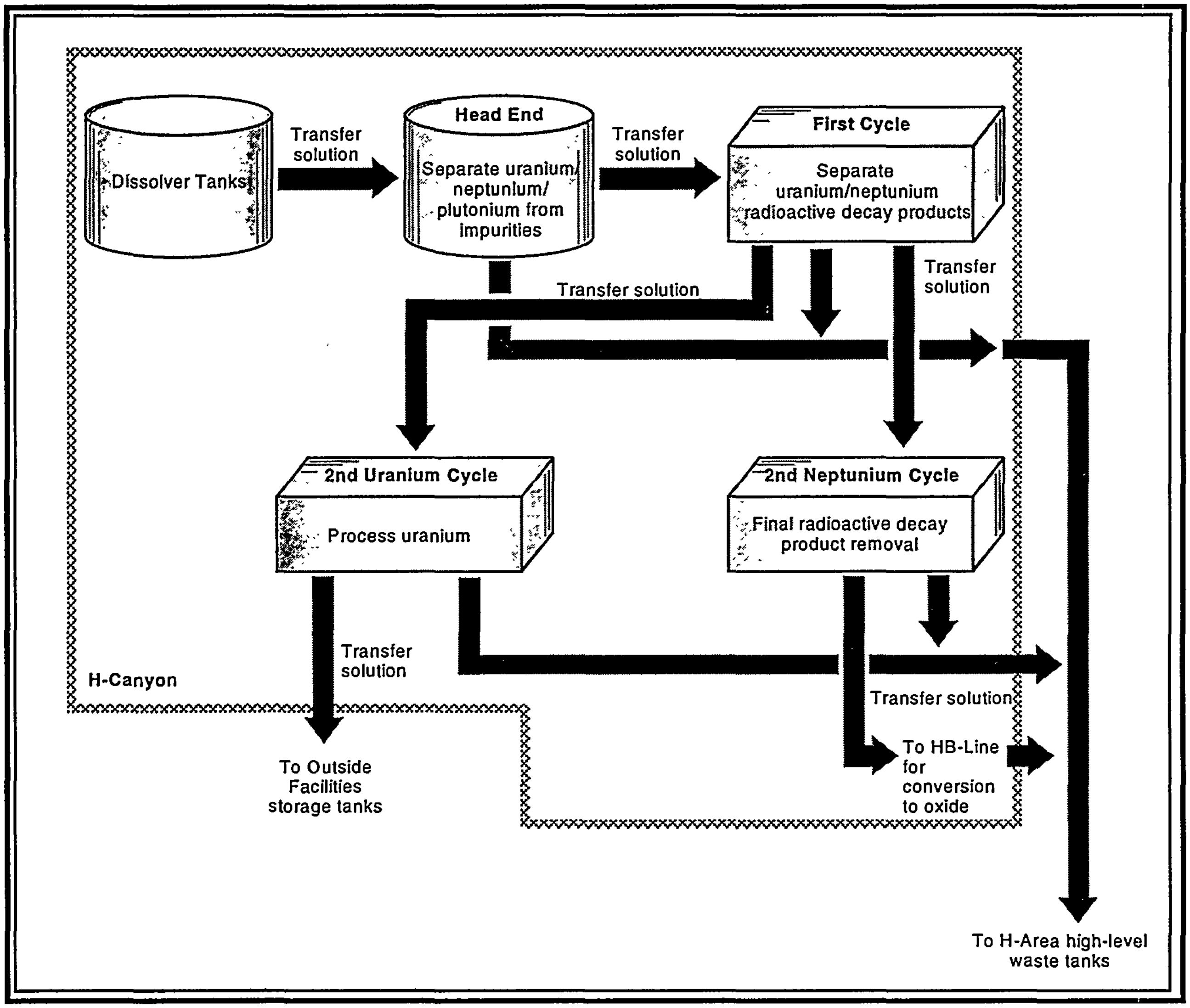


Figure C-10. Historic H-Canyon process flow.

The following paragraphs discuss major and support unit operations in H-Canyon:

- **Dissolution** - Irradiated reactor fuel on a rail car through an air lock is brought into the south end of the hot canyon. The fuel consists of highly enriched uranium fuel tubes clad in aluminum. As a result of the irradiation process, some of the material in the fuel was converted into radioactive decay products and other isotopes such as neptunium-237. Large water-filled casks on rail cars transport the fuel. The fuel is removed from the casks and loaded into a dissolver tank. Heated nitric acid and mercuric nitrates in the tank dissolve the fuel, resulting in a solution containing highly enriched uranium, neptunium, small quantities of plutonium, radioactive decay products from the reactor irradiation process, and the aluminum cladding. TE

- **Head End** - This process occurs in two steps to prepare the target solution for uranium and neptunium separation. First, gelatin is added to precipitate silica and other impurities. Then the solution is transferred to a centrifuge, where silica and other impurities are removed as waste. The clarified product solution is adjusted with nitric acid and water in preparation for the first cycle unit operation. The waste stream generated from the head end process is chemically neutralized and sent to the H-Area high-level waste tanks. The major components for this operation are a gelatin "strike" tank, a centrifuge feed tank, and a centrifuge.

- **First Cycle** - This operation, which occurs in the hot canyon, has two functions: (1) to remove radioactive decay products and other chemical impurities, and (2) to separate the solution into two product streams (highly enriched uranium and neptunium if recovery is scheduled) for further processing. During the solvent extraction process, the product solution passes through a series of mixer-settler banks. Before the introduction of the highly enriched uranium and neptunium feed solution, flows of solvent and acid (including nitric acid, as discussed for F-Area) solution start through the equipment. When equilibrium has been established, the feed solution from the head end is introduced. The chemical properties of the acid/solvent/feed solutions in contact with each other cause the radioactive decay products, the uranium, and the neptunium to separate. The first cycle produces four process streams: a highly enriched uranium solution with most of the radioactive decay product removed, which goes to the second uranium cycle; a neptunium solution with most of the radioactive decay products removed, which goes to the second neptunium cycle; a solvent stream, which goes to the solvent recovery system; and an aqueous acid stream containing most of the radioactive decay products and chemical salts used in the process, which goes to the high-level waste evaporators. If neptunium recovery is not desired, the solvent extraction cycle is revised and the neptunium is discarded with the aqueous acid stream. The equipment for this unit operation consists of mixer-settler banks, decanter tanks, and hold tanks.

- **Second Uranium Cycle** - The second uranium cycle (in the warm canyon) further purifies the highly enriched uranium solution from the first cycle and prepares it for transfer to the A-Line. The purification process is a solvent extraction process that occurs in a manner similar to that described for the first cycle. The highly enriched uranium product solution is transferred from the warm canyon to storage tanks in the A-Line facility, which is adjacent to the H-Canyon.

- **Second Neptunium (Second Product) Cycle** - The second neptunium cycle (in the warm canyon) purifies the neptunium solution from the first cycle if neptunium recovery is required by removing most of the residual radioactive decay products, and prepares the neptunium for transfer to HB-Line. The purification process is a solvent extraction process that occurs in a manner similar to that for the first cycle. The impurities are removed in an aqueous stream that goes to the low-activity waste unit operation for processing. The neptunium product solution is transferred to hold tanks for use as HB-Line feed material.

- **High- and Low-Activity Waste** - These unit operations reduce the volumes of the aqueous streams that contain radioactive decay products by using a series of evaporators in the hot and warm canyons. The feed to the evaporators originates with the primary separation process operations (e.g., the first cycle). The evaporator overheads, which contain most of the water and acid and very little of the radioactive decay product and chemicals used in solvent extraction, are transferred to tanks outside the building for acid recovery and recycling. The fission products and chemicals in the evaporator concentrate are neutralized and sent to the H-Area high-level waste tanks.

- **Solvent Recovery** - The primary purpose of this unit operation is to wash the solvent to remove impurities, and to recover and recycle the solvent extraction for reuse. The impurities are transferred to low-activity waste for processing. Solvent recovery is used with each extraction cycle.

C.4 FB-Line

The FB-Line is located on the top of the F-Canyon structure (see Figure C-11). Its exterior walls and roof are poured reinforced concrete. The portion of the structure that contains process equipment is approximately 39 meters (130 feet) long by 20 meters (67 feet) wide. The single-story extension to the north is about 11 meters (35 feet) wide by 6 meters (20 feet) long. Tanks and reaction vessels are enclosed in engineered cabinets or gloveboxes to minimize the spread of contamination and to provide shielding from radiation (see Figure C-12).

The FB-Line process includes purification and concentration of plutonium by cation exchange, precipitation of plutonium as a trifluoride, recovery of the trifluoride by filtration, drying of the trifluoride in an oxygen atmosphere, and reduction with calcium metal to form plutonium metal buttons. Figure C-13 shows the typical process flow through the line.

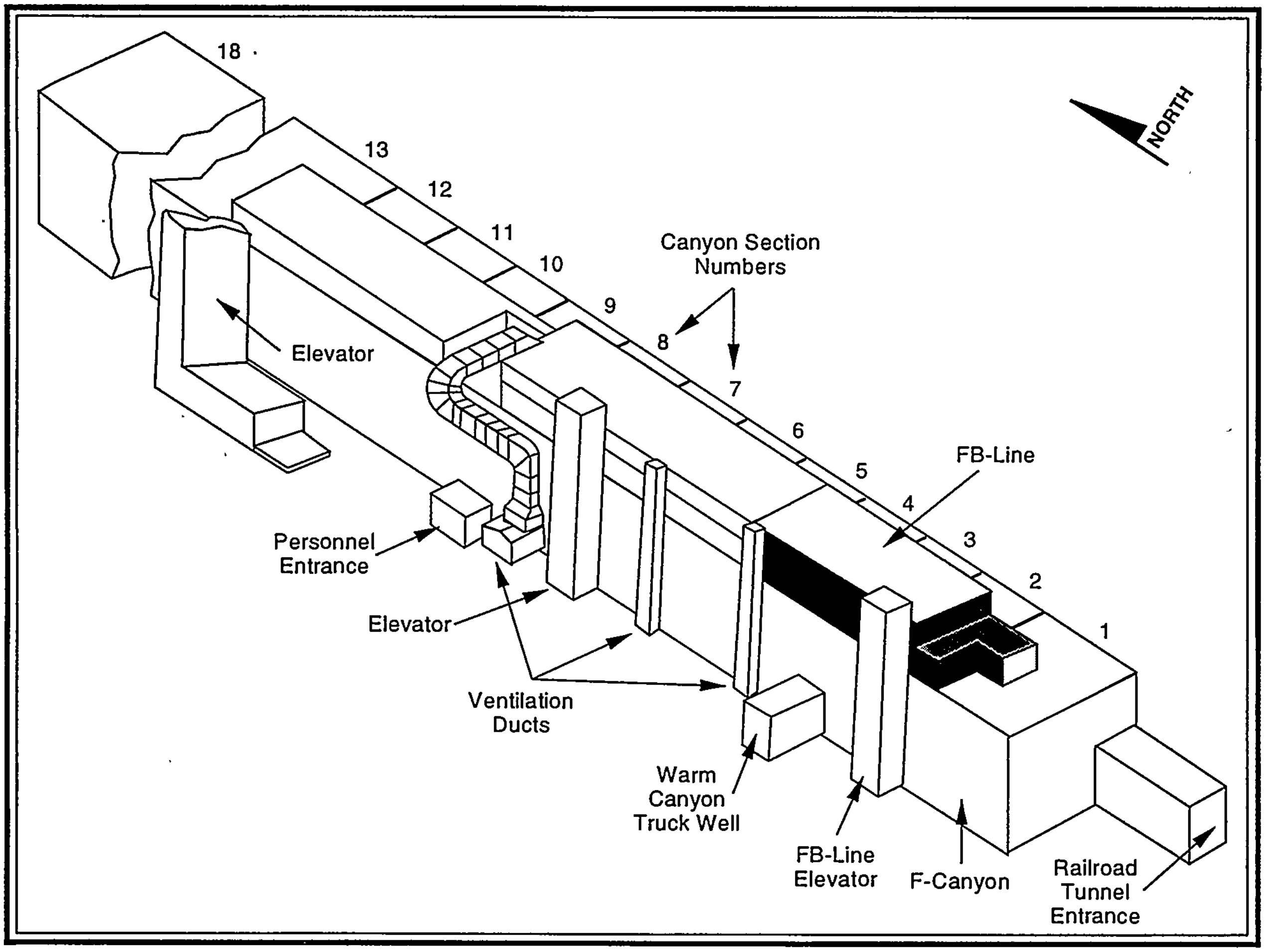


Figure C-11. F-Canyon and FB-Line facilities.

The following paragraphs describe the process steps and the FB-Line vaults:

- **Cation Exchange** - The purpose of the cation exchange step is to concentrate the diluted plutonium product solution from the PUREX process second plutonium cycle, remove metallic impurities, and further remove radioactive decay product impurities. This is accomplished by transferring the solution from the storage tanks in the warm canyon to cation exchange feed receipt tanks and then to the cation exchange columns in FB-Line. The plutonium from the feed solution is absorbed on the cation exchange resin. After the plutonium is absorbed on the resin, a strong acidic solution is used to flush the plutonium from the column. The solution containing the plutonium is filtered and transferred to a product hold tank for sampling and analysis, and then to a concentrate feed tank for subsequent precipitation.

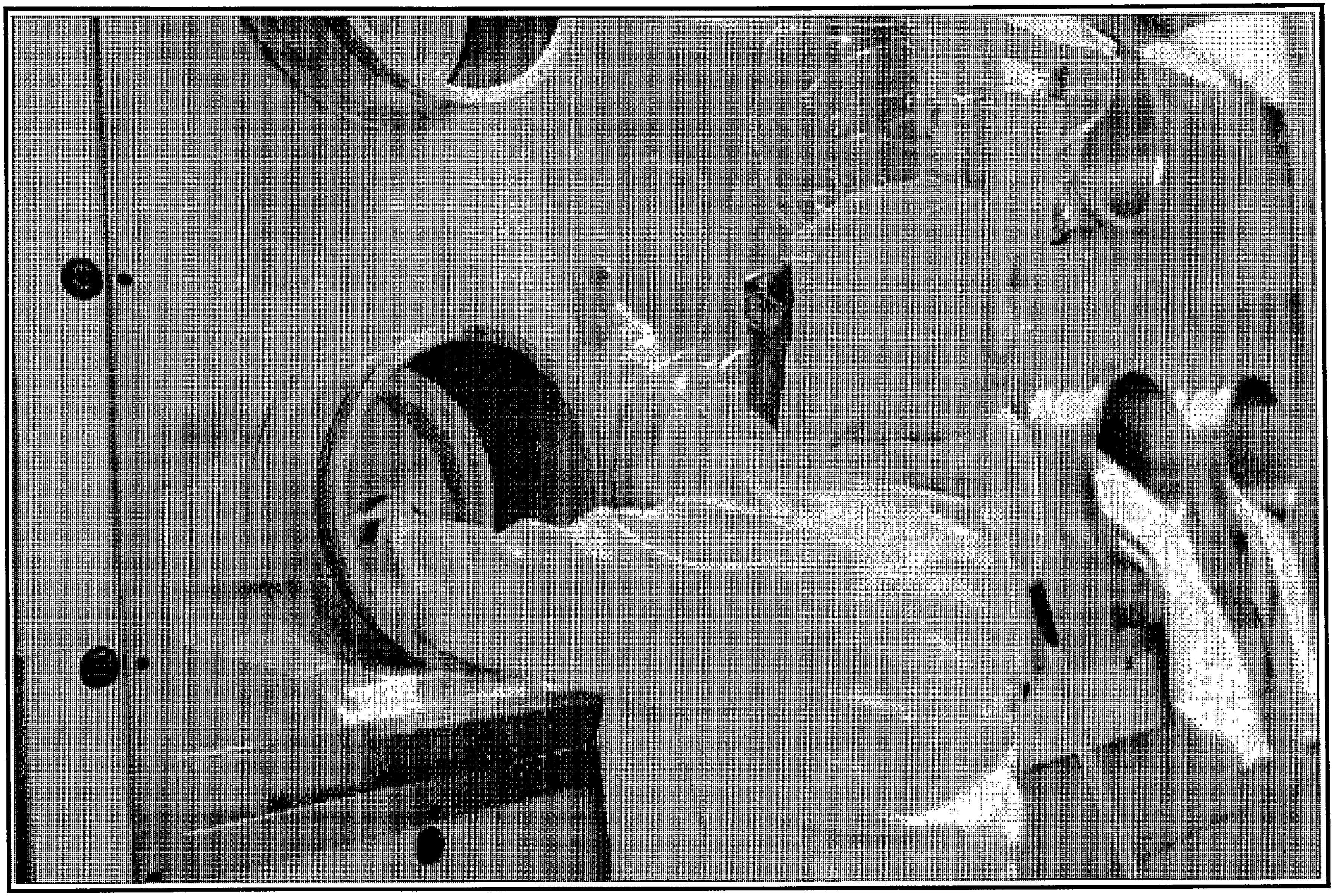


Figure C-12. FB-Line worker using a glovebox.

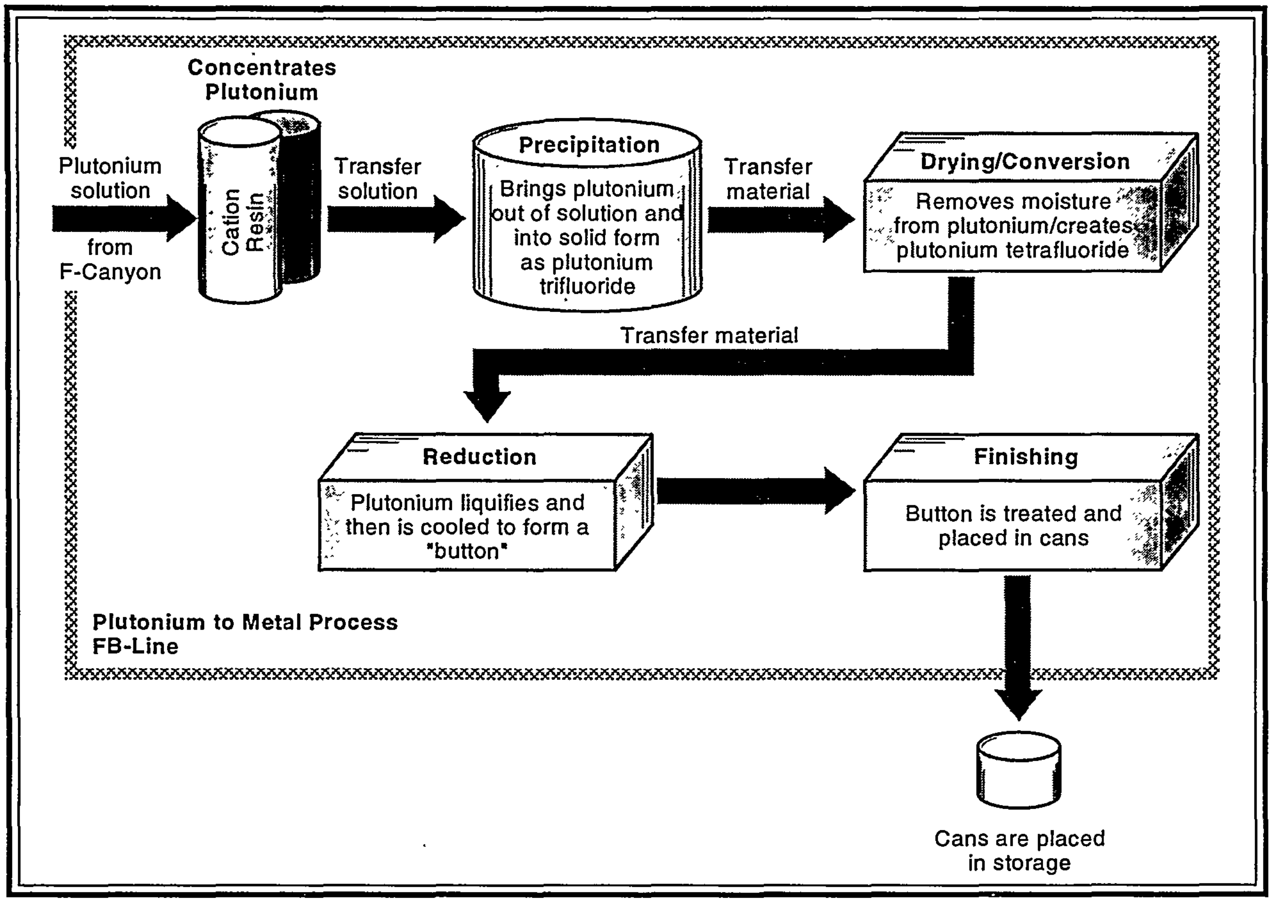


Figure C-13. FB-Line process flow.

The primary cation exchange equipment consists of 14 process tanks; 4 ion-exchange columns (which are shielded to reduce radiation levels); and 4 filters. This equipment is inside engineered cabinets at the FB-Line facility.

- **Precipitation and Filtration -** In this step, hydrofluoric acid is added to the plutonium solution from the cation exchange process. This action causes plutonium trifluoride to form and precipitate. The plutonium trifluoride precipitate is filtered; the result is a particulate cake. The plutonium trifluoride cake is then ready for the drying and conversion step. The 28 vessels associated with the precipitation and filtration process are in engineered gloveboxes in the FB-Line facility.

- **Drying and Conversion -** The actions that comprise the step occur in the FB-Line gloveboxes. The plutonium trifluoride of cake is transferred to the gloveboxes for drying and conversion to plutonium tetrafluoride. The cake is air-dried to remove residual moisture and then is placed in a conversion furnace. Residual water and other volatile materials evaporate at a low initial temperature; the vapors are drawn away by the vessel vent system. The temperature in the drying furnace is increased while the cake is blanketed in oxygen. This action converts the plutonium trifluoride to plutonium tetrafluoride powder.

- **Reduction** - The plutonium tetrafluoride powder from the conversion step is placed in a mixing and weighing vessel and weighed. The powder is then mixed with metallic calcium and placed in a prepared reduction vessel, which is 16.5 centimeters (6.5 inches) in diameter and 30.4 centimeters (2 inches) high. The material is heated to about 500°C (930°F), which initiates a chemical reaction that causes the plutonium powder to form molten plutonium metal. The heavier metal sinks to the bottom of the reduction vessel and forms a pool of molten plutonium.

  After the reduction vessel cools to the ambient temperature, it is opened and the solid plutonium, which has the appearance of a button, is removed. The reduction process is performed in the FB-Line gloveboxes.

- **Button Finishing** - An acid solution rinse removes exterior impurities from the plutonium button. Next the button is rinsed in water to remove the acid. After water rinsing, the button is air-dried, sampled, weighed, and placed in a tinned steel can, which is crimp-sealed. The can is removed from the process cabinet in a plastic bag and placed in a second tinned steel can, which is also crimp-sealed. This package is weighed and monitored for contamination and radiation. A leak test is performed and the canned buttons are removed from the gloveboxes and placed in a shipping or storage container and transferred to an FB-Line facility storage vault.

- **Storage Vaults** - Two FB-Line vaults store nuclear materials. The vaults were constructed with reinforced concrete floors, walls, and ceiling, and are equipped with storage racks. The nuclear materials are stored as described in Appendix A.

C.5 HB-Line

The HB-Line consists of the fifth and sixth levels of the H-Canyon. The HB-Line facility is approximately 12 meters (39 feet) high, 66 meters (215 feet) long, and 20 meters (67 feet) wide (Figure C-14). The HB-Line has reinforced concrete exterior walls and roof.

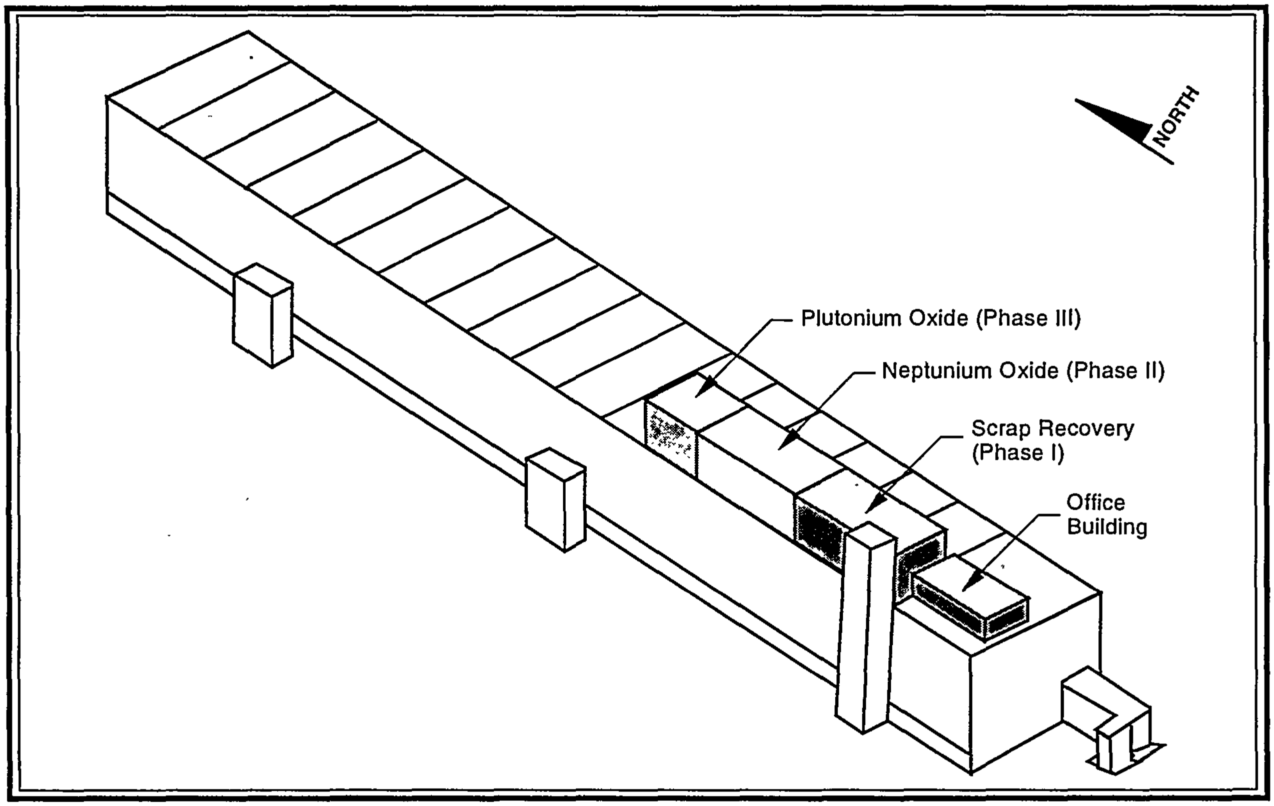


Figure C-14. H-Canyon and HB-Line facilities.

The facility has three primary processing areas (commonly referred to as "phases") and a vault for the storage of nuclear material:

- **Phase I** (Scrap Recovery) - Designed to dissolve scrap materials containing (1) various isotopes of plutonium (e.g., plutonium-238, -239, -242), (2) mixed oxides or alloys containing plutonium and enriched uranium, or (3) neptunium.

- **Phases II and III** (Oxide Lines) - Designed to convert solutions containing neptunium or plutonium to an oxide powder; Phase II is commonly used for neptunium and Phase III for plutonium-238 conversion.

The three phases contain (1) glovebox lines with process equipment, (2) instrument and electrical control rooms, (3) maintenance and equipment decontamination areas, (4) chemical feed preparation area, and (5) areas with air monitors, compressors, refrigeration units, and Halon fire protection systems.

The HB-Line vault is used primarily to store plutonium-238 oxide in stainless-steel containers. The containers are stored in a modular array under water, which removes heat generated from the radioactive decay of plutonium-238. The vault also has space for storing drums containing scrap metals, oxides, or compounds of plutonium-238, plutonium-239, neptunium-237, and uranium-235.

C.5.1 PHASE I

The Scrap Recovery phase generates nitrate solutions of plutonium or uranium-plutonium suitable for purification by anion exchange or solvent extraction in H-Canyon. Scrap materials containing plutonium or uranium-plutonium are placed in a charging container and manually added to the dissolver. The solid scrap is dissolved in nitric acid, transferred through a filter, collected in a tank, sampled, diluted with nitric acid, and transferred to a tank in H-Canyon as a nitrate solution. Duplicate lines in the recovery facility provide processing flexibility.

C.5.2 PHASE II

Phase II converts nitrate solutions of neptunium-237 to neptunium oxide (NpO_2) powder. The facility can also be used to convert nitrate solutions of plutonium to oxide. The neptunium nitrate solution, which has been purified in H-Canyon, is received in Phase II. It is purified by anion exchange, precipitated with oxalic acid, and calcined to produce oxide. The processing of plutonium through the facility uses the same unit operations.

C.5.3 PHASE III

Phase III converts plutonium-238 solution to plutonium-238 oxide. The plutonium-238 nitrate solution, which has been purified in H-Canyon, is received in Phase III. The plutonium is precipitated as plutonium oxalate, which is filtered and washed using inline filter boats. The filter boats containing the oxalate cake are transferred to the calcining furnace. The oxalate cake is calcined in an ultrapure oxygen-16 atmosphere to reduce neutron emissions from the oxide. The powder is loaded into a stainless-steel shipping container for storage or shipment off the Site. The Phase III conversion line has heavy neutron and gamma shielding to protect workers from the neutrons and gamma rays emitted by plutonium-238. Although the plutonium-238 oxide activity at the SRS is an ongoing NASA activity and not part of this EIS, this equipment could be used to stabilize some plutonium-238 residues that are not part of the NASA activity.

C.6 Outside Facilities (F- and H-Areas)

The Outside Facilities are adjacent to the canyon facilities and provide direct support to canyon processing operations. They do not normally have a housing containment. Figure C-15 shows the Outside Facilities in F-Area, which are similar to those in H-Area. Support operations include chemical storage, cold feed preparation, water handling, and acid recovery. The Outside Facilities also provide utilities, including water, electric power, and steam.

C.6.1 CHEMICAL STORAGE

The chemical storage facilities provide the receipt, bulk storage, and transfer of fresh liquid chemicals, which are sampled and analyzed before being accepted for storage. The storage tanks are stainless steel or carbon steel. During receiving operations, personnel obtain samples, verify proper connections and valve lineup, operate transfer pumps, and confirm that the solution enters the correct tank.

The building consists of an enclosed storage area and a mixing area with open sides. Access roads surround the building and a railroad spur is on one side. The grounds also include two tank truck stations, a truck dock, a railroad dock, and a small hydrazine mononitrate storage area.

Stored chemicals are pumped from the storage facilities to points of use in buildings. Smaller quantities are distributed to other parts of the plant through a drum loading and dumpster filling station. Organic solvents and caustics are pumped directly from their respective storage tanks.

C.6.2 WATER HANDLING FACILITIES

The water handling facilities receive and store condensed water from steam condensates originating in the acid recovery unit reboiler and general-purpose evaporator heaters. In addition, deionized water from the powerhouse is also received and stored for use as process water. This water is recycled at the Effluent Treatment Facility and used to provide process water and acidified water streams for the canyons. Some water is discharged to Fourmile Branch after treatment. Tanks in this facility are also used to retain water pending analysis to permit disposal or reevaporation, if necessary.

The cooling water system provides cooling water for the hot and warm canyon process vessels. Underground pipes carry water to the canyons, where it is distributed. The water passes through coils

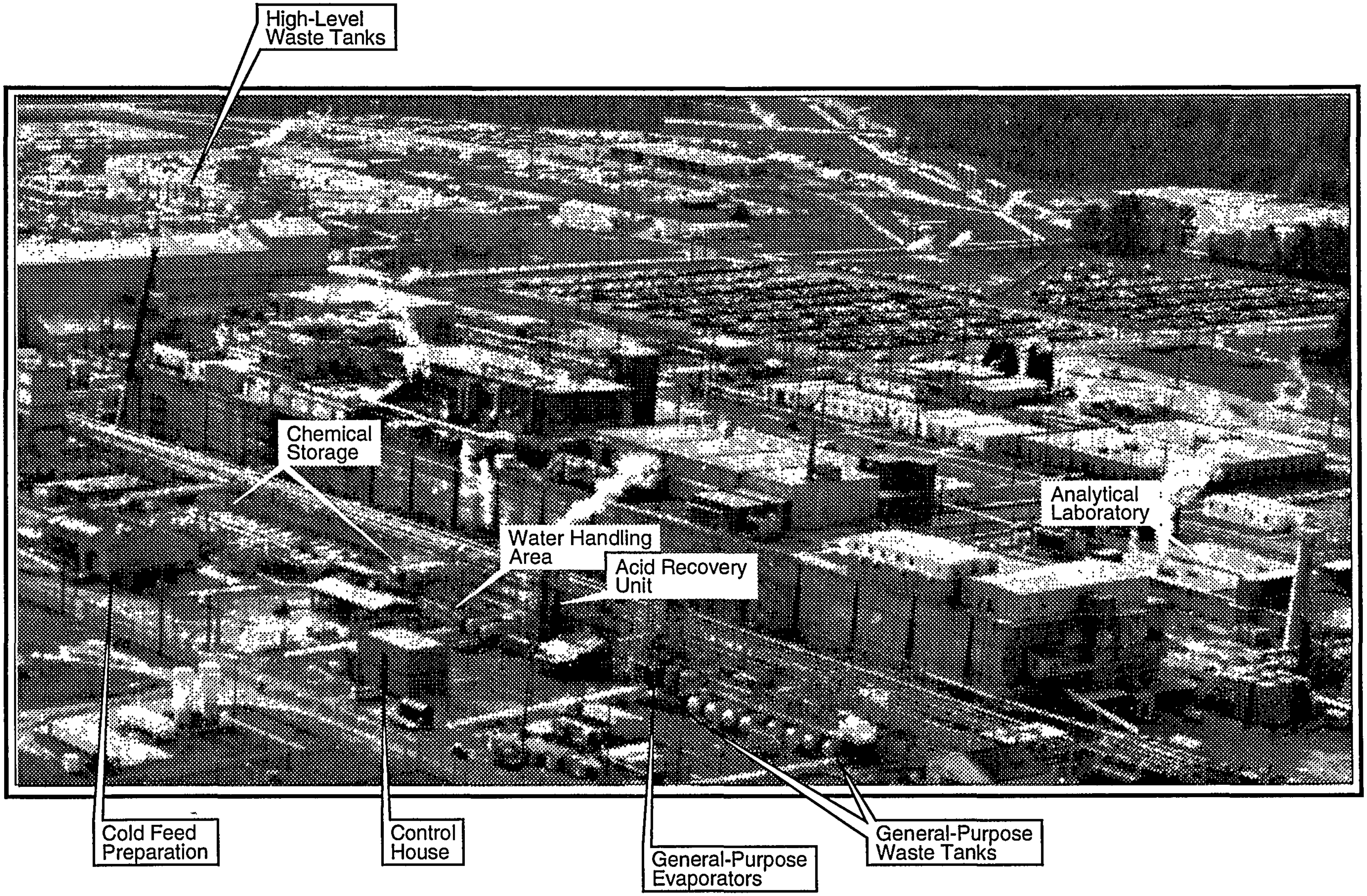


Figure C-15. F-Area, showing Outside Facilities (view toward southeast).

inside the vessels and then flows back out of the canyon. Constant monitoring detects radioactivity in the water in the event of a cooling coil leak. If radioactivity is detected, the water is diverted to a treatment facility where the radioactivity is reduced below applicable limits before the water is discharged.

The primary equipment for water handling consists of hold tanks, skimmer tanks, and heat exchangers. Hold tanks are mounted on concrete saddles in shallow pits (concrete pads) that drain to a sump. Heat exchangers and skimmer tanks are rack-mounted.

C.6.3 ACID RECOVERY UNIT

The Acid Recovery Unit concentrates nitric acid condensates for reuse. The condensate comes from such acid evaporation processes as high- and low-activity waste unit operations in the canyons. Each acid recovery unit is a distillation column that has a straight shell height of 8 meters (27 feet) and an outside diameter of 2 meters (6.5 feet). A reboiler is attached to the bottom side of the column. The Acid Recovery Unit Feed Tank receives condensates from the high-activity waste continuous evaporators; the low-activity waste condensate is brought in directly from the continuous evaporator. Canyon samples or in-line monitors determine if there is radioactivity in the condensate before it reaches the feed tank or distillation column. After recovery, the concentrated acid is pumped to a storage tank for transfer to the canyon as required.

C.6.4 GENERAL-PURPOSE WASTE TANKS

This facility consists of eight storage tanks grouped in two sets of two and one set of four to collect various aqueous wastes with low contamination levels. The first set collects solutions from various sumps and catch tanks in the canyons. (Highly contaminated liquids in the hot and warm canyons are processed with equipment in the hot canyon and are not part of this system.) The second set collects wastes from chemical storage tank areas. The third set collects wastewater from sumps and pits in the Outside Facilities. The aqueous waste in these tanks is transferred to other areas in the Outside Facilities (e.g., the general-purpose evaporator) for processing.

C.6.5 GENERAL-PURPOSE EVAPORATOR

General-purpose evaporators concentrate aqueous waste (principally from the general-purpose waste tanks) that have radioactivity levels higher than disposal limits, but are low enough to enable evaporation in unshielded equipment. Each stainless-steel evaporator has a straight shell height of

4.8 meters (16 feet) and an outside diameter of 1.8 meters (6 feet). The evaporators are operated as flash evaporators with forced bottoms circulation. Evaporator bottoms are concentrated and retained for analysis before their discharge to high-level waste tanks.

C.6.6 WASTE HANDLING FACILITIES

The waste handling facilities are tanks used for the storage and transfer of high- and low-activity wastes, primarily from F-Area and Savannah River Technology Center laboratory facilities. Low-level wastes are transferred to the General-Purpose Evaporator for processing. High-level wastes are transferred to the laboratory waste evaporator in F-Canyon. The waste handling vessels and cells are enclosed by a concrete vault, which has a sloped floor and sump to collect leakage and a ventilation system consisting of two air heaters, eight roughing filters, eight high-efficiency particulate air (HEPA) filters, four dampers, and two exhausts.

C.7 A-Line (F-Area)

The A-Line Facility is about 50 meters (150 feet) from the southeast end of F-Canyon (see Figure C-5). A-Line is a three-story structure, with a fourth-story penthouse at its north end, and a basement. Overall, A-Line is 15 meters (49 feet) wide and 40 meters (135 feet) long. The foundation and basement walls are reinforced concrete. Exterior walls are reinforced concrete framing with flat cement asbestos board supported on a steel framework. Principal equipment includes evaporation units to concentrate uranyl nitrate solutions, continuous denigrators to produce an oxide powder, handling equipment to package the oxide powder, and equipment to recover acid fumes from the conversion process.

The primary purpose of the facility is to convert depleted uranium nitrate solutions to uranium trioxide (EAU) powder. Dilute uranium solutions from the F-Canyon PUREX process are transferred through pipes to A-Line tanks. Initial concentration takes place in two evaporation stages. After concentration, the resulting uranyl nitrate hexahydrate solutions are transferred to one of three continuous denigrators. The uranyl nitrate hexahydrate solution is fed to the continuous denitrator, which operates at approximately 260°C (500°F). The water fraction of the uranium solution is flash-evaporated and the uranyl nitrate is deposited on the outer surface of the uranium oxide in the denitrator. The uranyl nitrate is thermally decomposed at the denitrator temperature and forms spherical uranium oxide particles. The uranium trioxide is transferred through a pneumatic line to hoppers where the oxide is discharged into 208-liter (55-gallon) drums for storage.

C.8 A-Line (H-Area)

The H-Area A-Line receives a dilute aqueous solution of uranyl nitrate enriched in uranium-235 from the H-Canyon. The A-Line building is an open I-beam structure that supports tanks, piping, valves, pumps, and walkways. The structure is built on a concrete foundation near the southeast corner of H-Canyon (it is behind H-Canyon as shown on Figure C-6).

Six stainless-steel storage and loading tanks are associated with A-Line. Five are inside the A-Line structure; the sixth, a transfer tank, is beside the canyon.

A large tank in the H-Area Outside Facilities area connects the A-Line to the Uranium Solidification Facility. The uranyl nitrate solution from the A-Line can be transferred to this tank, called the Uranyl Nitrate (UN) Storage Facility, for storage until the solution is processed to an oxide. The primary process consists of the transfer of uranyl nitrate product solution from the canyon through a series of tanks for storage. The solution is decanted in a warm canyon tank and sampled for analysis at the control laboratory. When analytical results confirm that the solution is within specifications, the solution goes to A-Line.

TE

C.9 Receiving Basin for Offsite Fuel

The Receiving Basin for Offsite Fuel (RBOF) in H-Area (which is shown in Figure C-16) provides interim storage of irradiated spent nuclear fuel elements from SRS reactors and domestic and foreign test and research reactors. RBOF has been operating and receiving offsite fuels since 1964. The facility is approximately 42 meters (139 feet) long, 45 meters (148 feet) wide, and 14 meters (45 feet) high. The building superstructure is fabricated from standard structural steel members with walls of transite and concrete block. The roof is of built-up composition supported by long-span steel joists.

TE The Receiving Basin for Offsite Fuel has an unloading basin, two storage basins, a repackaging basin, a disassembly basin, and an inspection basin, all underwater. Fuel is handled or stored beneath at least 3 meters (10 feet) of water to provide shielding against radiation. The reinforced-concrete basins are below grade. They have either chemical coatings or stainless-steel linings for ease of decontamination. The storage lattice in the basins consists of rows of racks of aluminum I-beams. Gratings, guide plates, and spacers between the racks separate individual storage positions and provide the spacing required for criticality safety.

In addition to the water-filled basins, the Receiving Basin for Offsite Fuel has a receiving bay, dry TE cask inspection pit, control room, office areas, equipment storage areas, and concrete cells containing tanks for water decontamination (deionization) and temporary storage of radioactive liquid waste.

The facility has a 91-metric-ton (100-ton) bridge crane that travels on rails approximately 9 meters (31 feet) above grade. The crane has two 45-metric-ton (50-ton) hoists and two 2.7-metric-ton (3-ton) hoists. The crane supports travel over the cask receiving, unloading, and fuel storage areas of the facility.

The receiving bay on the north side of the Receiving Basin for Offsite Fuel receives shipping casks containing irradiated fuel delivered by either truck or rail. Radiological surveys are performed on TE the casks to determine any external radiation and surface contamination levels. The casks are transferred to a pit where steam is used to remove any dirt, debris, or radioactive contamination. After cleaning, the cask is vented and filled with water, and the water is sampled to detect TC contamination that would indicate fuel that might have become damaged or "failed" during transit. The cask lid bolts are loosened and the cask is transferred to the cask basin using the 91-metric-ton (100-ton) overhead crane. The cask is lowered into the cask basin until the top of the lid is approximately 0.9 meter (3 feet) above the water surface and the lid bolts are removed. The cask is lowered to the bottom of the basin and the lid is removed. Fuel elements are removed from the cask and placed in transfer buckets, cans or bundles, depending on the fuel design. The bucket, can, or bundle is moved through an underwater door into a canal system that links the storage basins. The bucket, can, or bundle is placed in a storage rack and the process is repeated until all fuel has been unloaded from the cask.

Separate basins are available in the Receiving Basin for Offsite Fuel for segregation of damaged or TE failed fuel, for disassembly of fuel components by mechanical means (such as cutting), or inspection and measurement. The basin water is circulated through a filter and then a deionizer for purification and clarification. Filters or deionizers are replaced periodically; the frequency depends on the radioactivity or impurity levels in the water.

C.10 Laboratories

SRS laboratories include the Savannah River Technology Center (SRTC) and smaller facilities.

C.10.1 SAVANNAH RIVER TECHNOLOGY CENTER

The Savannah River Technology Center is in A-Area (see Figure 1-2) near the northwest boundary of the SRS. Building 773-A is the Main Laboratory Building. SRTC activities include research and development in the production and processing of actinides, waste processing, hydrogen technology, fuel technology, interim waste technology, robotics, and fabrication technology. Building 773-A consists of six sections on two levels. The second level is used for offices; the first level contains nearly 100 small laboratory areas, which include laboratories that handle radioactive materials, radiochemical laboratories, equipment for cold (nonradioactive) research and development in the vitrification process used in the Defense Waste Processing Facility, and equipment used in programs to encapsulate the isotope californium-252 for use in the treatment of some cancers.

C.10.2 OTHER LABORATORIES

The Central Analytical Laboratory facility (see Figure C-15) consists of Buildings 772-F and 772-1F. The primary activities of this laboratory involve analyses on small samples of radioactive materials from various facilities. A wide range of analytical measurements, including those required for process control, accountability, and product specification analysis can be performed in this laboratory.

The 322-M Metallurgical Laboratory (see Figure C-2) is a metallurgical and physical testing laboratory, which originally supported the reactors and reactor materials fabrication facilities. At present, it supports the decontamination of Building 321-M and performs testing and support services such as weld testing, corrosion testing, material strength testing, and chemical cleaning for a number of SRS facilities, including the Defense Waste Processing Facility.

The TNX laboratory is a pilot-scale testing laboratory near the southwest SRS boundary (see Figure 1-2). DOE used the laboratory to test Separations Area equipment. More recently, it had been used to support the Defense Waste Processing Facility. At the present time, it contains some depleted uranium previously used in testing related separations activities.

C.11 Building 235-F

Building 235-F (see Figure C-5) is a reinforced concrete structure, approximately 68 meters (220 feet) long, 33 meters (110 feet) wide, and 9 meters (30 feet) high. The exterior walls of the two-story structure are 36 centimeters (14 inches) thick. This building houses materials storage vaults that are in use and process areas that are no longer in use. The inactive process areas are the Actinide Billet (AB) Line, the Plutonium Experimental Facility, and the Plutonium Fuel Form Facility. The building also contains the Metallography Laboratory.

In the Actinide Billet Line, neptunium oxide powder was received from HB-Line, blended with aluminum powder and pressed into compacts that were assembled into aluminum-clad billets and welded. The fabricated billets were transferred to M-Area where they were extruded through presses into target tubes for irradiation in SRS reactors. The operations in the Actinide Billet Line included receiving, weighing, blending, die preparation, cold pressing, loading compacts in billets, welding, and leak testing. The Actinide Billet Line was constructed in 1961 and last operated in 1983.

The Plutonium Experimental Facility was used for pilot-scale demonstration of manufacturing processes for encapsulated plutonium-238 pellets. After successful demonstration in this facility, the process was installed on a production scale in the Plutonium Fuel Form facility. DOE built the Plutonium Experimental Facility in the early 1970s and operated it until 1982.

The Plutonium Fuel Facility was used to compress plutonium-238 oxide into small pellets and to encapsulate them in an iridium cladding. The encapsulated pellets served as General-Purpose Heat Sources for a variety of applications. Supporting operations included weighing, analyzing, welding, leak testing, and packaging the pellets for shipment. DOE built this facility in 1977 and operated it until 1983.

DOE built the Metallography Laboratory in 1989 to replace an obsolete laboratory in Building 235-F that performed metallurgical examination of encapsulated plutonium-238 heat sources to verify process quality control. The facility has never operated.

Three storage vaults in Building 235-F are used to store nuclear materials. These vaults provide storage for a wide variety of materials in approved shipping containers, other metal containers, scrap containers, and those requiring special cooling due to significant decay heat. (See Appendix A for additional information.)

C.12 Depleted Uranium Storage Facilities

SRS warehouses and storage facilities in F-, N-, R-, and G-Areas, contain depleted uranium. These facilities are of conventional construction (i.e., steel framed with concrete pads and metal siding).

C.13 Building 247-F Vault

DOE used Building 247-F (see Figure C-5) to produce highly enriched uranium fuel for use in naval reactors. DOE has deinventoried the process areas of the facility, and only residual levels of nuclear material remain.

The only portion of Building 247-F that DOE could use to implement any of the alternatives evaluated in this EIS would be the vault for storage of nuclear material in containers designed for storage or transport. The floor, walls, and ceiling of the vault are of reinforced concrete. The vault was designed to resist high winds, earthquakes, and other natural phenomena. DOE used the vault to store various forms of highly enriched uranium in cans, drums, and cylinders.

The only other areas of Building 247-F that could be used would be the loading dock and the corridor that leads to the vault. These areas would be used during the receipt or transfer of material. The receiving dock would be used during the loading or unloading of truck shipments to other SRS facilities. The corridor leading to the vault would be a staging area to inventory the packages for accountability or for temporary segregation of a damaged container. Nuclear material would not be removed from the storage containers in Building 247-F. If repackaging were required, the container would be transferred to another SRS facility that had the appropriate equipment (e.g., gloveboxes) for opening.

C.14 New Facilities

The facilities described in the following sections either do not exist or have not been completed. Several alternatives for the stabilization or processing of nuclear materials to a storable or transportable form would require the use of capabilities that these facilities would provide.

DOE has evaluated the actions required for the modification and construction described in this section (with the exception of the Dry Storage Facility). Chapters 4 and 5 of this EIS describe these impacts.

C.14.1 URANIUM SOLIDIFICATION FACILITY

The Uranium Solidification Facility (USF) would convert liquid uranyl nitrate highly enriched in uranium-235 to uranium oxide powder for long-term storage. This facility would be inside H-Canyon, covering three levels at the south end. It would be in space previously used for the processing of plutonium-238 (the "old" HB-Line). The construction of the USF began in 1989 and was suspended in late 1993 after reaching 80-percent completion. The completion and operation of this facility is one of the alternatives considered in this EIS for the stabilization of materials containing highly enriched uranium. The USF is separated from the rest of the canyon areas by concrete walls and slabs. Structural steel framing is used for the platforms that contain processing equipment. The foundation of the H-Canyon building supports this facility.

The Uranium Solidification Facility would consist of six primary unit operations: (1) uranyl nitrate receipt and feed, (2) thermosiphon evaporation, (3) wiped-film evaporation, (4) denitration, (5) uranium trioxide product handling, and (6) condensate collection and concentrate recycle. Dilute uranyl nitrate solutions would be transferred from H-Canyon to the USF and concentrated using two thermosiphon evaporators. The uranyl nitrate solution would be further concentrated using a wiped-film evaporator. The solution would be fed into a paddle-type denitrator for conversion to uranium trioxide. The product from the USF would be weighed, analyzed for uranium-235 content, placed in shipping containers, and stored in a vault. Fumes from the thermosiphon evaporators would be collected in the condensate collection tanks. Fumes from the wiped-film evaporator and denitrator would be collected in the concentrate recycle tanks. The solution in these tanks would be analyzed and transferred back to H-Canyon.

The Uranium Solidification Facility would have four processing areas. The first area would contain a product glovebox room, a process area, an electrical control room, and an analytical laboratory. The product glovebox room would contain a uranium oxide product glovebox. The process area would include the following equipment:

- Primary evaporator feed tank module
- Primary evaporator module
- Secondary evaporator feed tank module
- Secondary evaporator module
- Wiped-film evaporator feed tank module
- Two condensate collection tank modules
- Concentrate recycle collection module
- Vessel vent drain collection module
- Primary and secondary evaporator control cabinets
- Feed tank control cabinet
- Dumbwaiter for transfer of waste between levels

The electrical control room would house the electrical power distribution equipment for both normal and standby power requirements for the Uranium Solidification Facility. The analytical laboratory would have six analytical cabinets with ventilation hoods, gloveboxes, a demineralizer, and a lab waste collection tank.

The second processing area would include the denitrator room and the process area. The denitrator room would contain the denitrator and support equipment. The process area would contain the primary and secondary evaporator feed tanks.

The third processing area would include the facility control room and maintenance access. The control room would contain a control panel to annunciate general alarms from the facility for both the process and support equipment. This area would enable maintenance access to the process equipment modules.

The fourth processing area would include the product vault, other support services, the product monitoring room, and the mechanical equipment rooms. The product monitoring room would be inside the product vault. The mechanical equipment rooms would contain a vessel vent high-efficiency mist eliminator filter module, a vessel vent high-efficiency particulate air filter module, the glovebox HEPA filter housing, the enclosure HEPA filter housing, glovebox exhaust fans, process enclosure exhaust fans, and low-volume air monitoring exhaust fans.

C.14.2 DRY STORAGE FACILITIES

TC

DOE evaluated ways to accelerate the time needed to provide a Dry Storage Facility at the SRS, and concluded it might be possible to provide such a facility in about 5 years and to accomplish fuel and target transfer operations in about 3 years. DOE made several assumptions to accelerate the design and construction schedule, specifically an emergency funding authorization to achieve a Fiscal Year 1998 budget line item and simultaneous approval of engineering and construction design specifications (i.e., Title 1 and 2 design). In addition, DOE assumed an integrated approach to design and construction proposal packages. The operation duration (3 years) was derived without regard for L-Reactor basin facility operations that could be required (e.g., foreign research reactor fuel or domestic research reactor fuel unloading or handling operations) or for interim target or spent nuclear fuel inspection activities that could be required to evaluate or qualify the techniques used to dry store the failed targets or fuel.

C.14.2.1 Spent Nuclear Fuel Storage Using a Dry Vault

An aboveground dry vault (Modular Dry Storage Vault) is a self-contained concrete structure that would enable dry spent nuclear fuel handling and storage. This design represents an integrated spent nuclear fuel storage approach and would consist of four major components: a receiving loading/ inspection area, spent nuclear fuel storage canisters, a shielded canister handling machine, and a modular array for storing the spent nuclear fuel storage canisters. The receiving area would use a wet pool for unloading the casks and for short-term (1 to 3 years) storage of spent nuclear fuel elements with a heat load that exceeded 40 Watts per element. The vault would consist of several modular units, and each unit could provide storage for hundreds of spent nuclear fuel assemblies. The vault itself would contain a charge/discharge bay with a handling machine above a floor containing steel tubes that would house the (removable) spent fuel canisters. The bay would be shielded from the stored fuel by the thick concrete floor and shield plugs inserted in the top of the steel storage tubes. The steel tubes, which would serve as secondary containment for the fuel, would descend into an open storage area. Large labyrinth air supply ducts and discharge chimneys would permit natural convection cooling of the tubes, while the perimeter concrete walls would provide shielding. The design would enable expansion by the addition of units of arrays to the end of the vault or by construction of another vault. The vault facility would also include a receiving and loading bay that would allow handling of shielded shipping casks and unloading of spent nuclear fuel into the short-term wet storage pool. The receiving bay would enable fuel inspection and canning as required, and could be used for fuel characterization with additional equipment and modifications. Although the physical condition of the spent nuclear fuel elements probably would not require extensive canning, the design would include the capability to can the entire spent fuel inventory. TC

In operation, the shipping cask would be lifted by a crane and placed in the unloading area of the small wet pool. The fuel elements would be removed underwater, and examined; if the heat generation rate was below 40 Watts per element, the fuel would be placed in the transfer canister. The transfer canister would be subsequently drained, dried, and seal-welded. The handling machine then would place the fuel inside the spent nuclear fuel storage canister, and would transport the loaded canister to the storage tubes. The handling machine would include radiation shielding. Heat dissipation would occur by natural convection from the surfaces of the handling machine and canister. Decay heat would be dissipated by natural convection; air would enter through inlet ducts at the bottom of the vault module, pass around the outside of the steel storage tubes containing the spent fuel canisters, and exit through outlet ducts at the top of the module. Therefore, the vault would be a complete integrated facility with all of the required capabilities for spent nuclear fuel handling and storage.

The vault facility would store spent fuel in canisters that are approximately 40.6 centimeters in diameter by 4.6 meters (16 inches by 15 feet) long. As currently envisioned, spent fuel would be stored in the canister in five levels with four elements per level, for a total of 20 fuel elements per canister. The vault design would allow 36 to 44 canisters per array unit, depending on the decay heat of the spent fuel and a cladding temperature limit nominally 175°C (347°F) for aluminum cladding with an air inlet temperature of 49°C (120.2°F). Thus, the number of vault units and arrays required for the storage of elements with a decay heat between 10 and 40 Watts per element would be 27.

Most of the spent nuclear fuel is likely to have decay heats between 10 and 40 Watts per element. For "cold" fuel (less than 10 Watts per element), more than 44 spent nuclear fuel canisters could be placed in a vault unit. However, this would require a customized design, which could unnecessarily increase costs and implementation time. Figure C-16 shows the layout of the modular vault dry storage facility (10 to 40 Watt element basis).

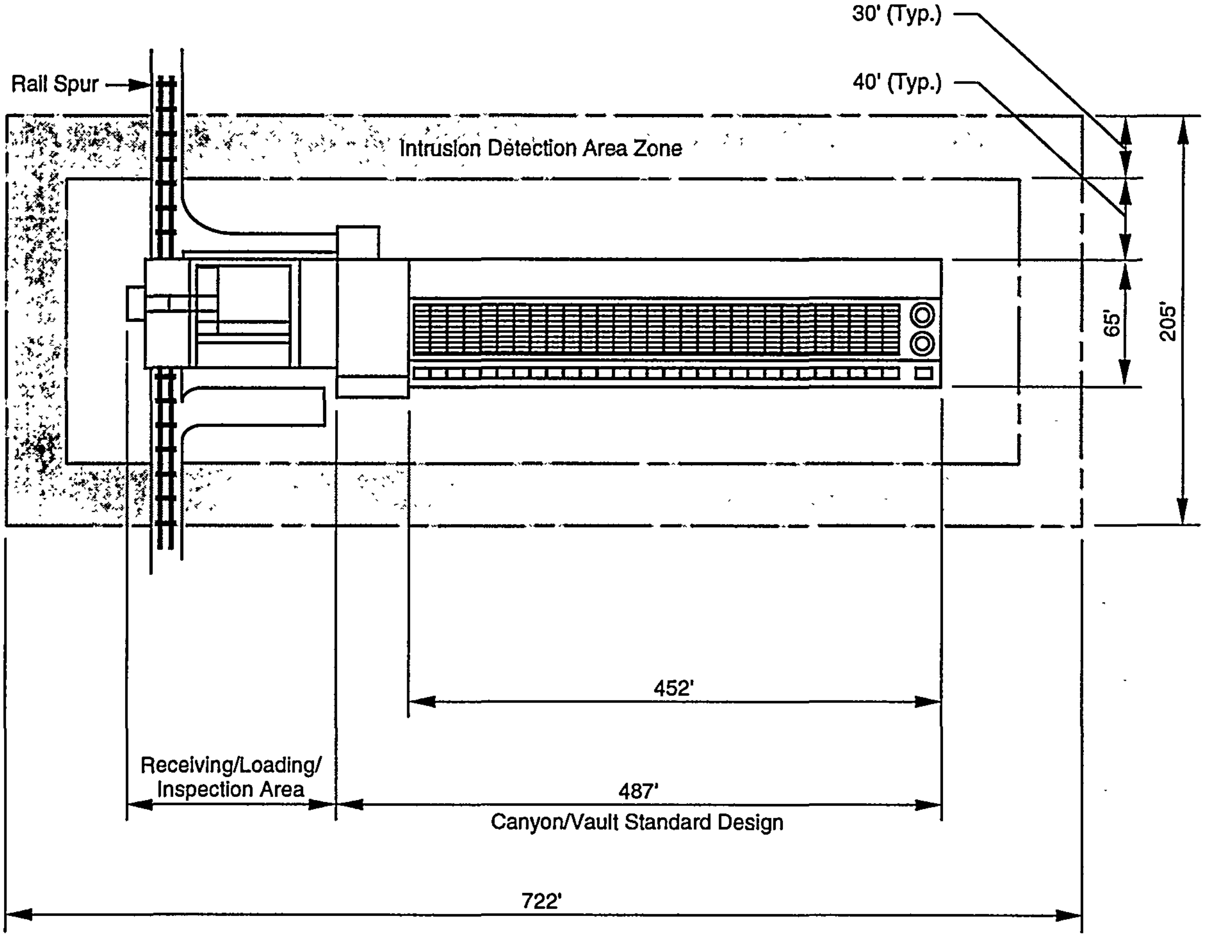
TC Criticality concerns would be addressed primarily by the tube spacing in the vault. Borated concrete could also be used. For spent nuclear fuel, criticality probably would not be a significant concern because a considerable fraction of the fissile uranium would have been consumed and neutron-absorbing fission products would be present.

The Nuclear Regulatory Commission has licensed this vault design, without a pool, for the Fort St. Vrain nuclear power plant site. The design represents a complete standalone facility that could be dedicated to spent nuclear fuel without requiring the utilization of any other facilities. Cask handling, fuel transfer to a canister, and fuel storage could be accomplished in the facility. Additional facilities or modifications to the inspection area, including a pool, would be required for fuel characterization.

C.14.2.2 Spent Nuclear Fuel Storage Using Dry Casks

Dry cask storage would include the use of concrete casks, both vertical and horizontal, metal casks, and multipurpose casks, and would consist of the following components:

- A staging facility for cask receipt and unloading and for loading spent fuel into the dry storage casks. The staging facility would have a wet pool for unloading the casks and for short-term (1 to 3 years) storage of spent nuclear fuel with a heat load that exceeded 40 Watts



10-40 W/Element Scenario
25,000 FRR spent fuel element modular array (nominal) storage required (29 modules with 44 tubes per module - 20 spent fuel elements per tube)
• Passive cooling • Monitored storage canisters • Retrievable for periodic inspection • Self-contained facility for staging and storage of spent fuel elements

Note: All dimensions shown are approximate and should be used only for a conceptual "footprint" estimate.

TC

Figure C-16. Layout of a modular vault Dry Storage Facility for spent nuclear fuel.

per element. This facility would include capabilities for drying the spent fuel and canister, inserting the spent fuel and canister with helium or nitrogen, and welding the storage canister closed.

- An inspection and characterization facility for examining fuel integrity and for canning leaking fuel as required. This facility could be incorporated in the staging facility (as an inspection cell) or placed immediately adjacent to it. Although the physical condition of the spent reactor elements probably would not require extensive canning, the design would provide the capability to can the entire spent fuel inventory.

- A dry storage cask (usually concrete) for the shielding and the structural stability of the spent nuclear fuel storage. The Multipurpose Canister currently under development could be used.

- A transfer mechanism, such as a dedicated truck/trailer combination with a ram for horizontal modules or a crane for vertical modules.

- A separate spent nuclear fuel canister could be included. It would typically be approximately 4.6 meters (15 feet) long and 1.7 meters (5.5 feet) in diameter, and would weigh approximately 33 metric tons (36 tons). TC

The dry cask approach would require the staging facility to receive and inspect the fuel shipment. The shipping cask would be unloaded in a small wet pool in the facility. Subsequently, fuel would be loaded into the dry cask (or fuel canister for the horizontal cask) and the cask would be placed on an outdoor concrete slab. The horizontal approach would use a dry fuel transfer canister to contain the fuel. This canister would be placed in a shielded transfer cask and moved to the outside modular storage facility. A hydraulic ram would insert the transfer canister in the horizontal storage module and seal it with a shield plug. Thus, dry cask storage would always rely on the use of another facility.

Dry storage casks would be designed to withstand normal loads and the effects of design-basis accidents such as earthquakes, tornadoes, and floods. Concrete would provide radiation shielding for gamma rays and neutrons. Natural air circulation would dissipate the heat; air would enter through inlet vents near the bottom of the cask, pass around the spent nuclear fuel canister, and exit near the top. Screens and grills would keep birds and animals out of the cooling duct area.

The application of dry cask storage technology to spent nuclear fuel would depend on the heat load. Horizontal casks are likely to be slightly more restrictive than vertical casks with respect to the heat load and are thus the focus of attention. The standard design for a horizontal fuel canister would provide 24 or 52 sleeves (i.e., pressurized water reactor or boiling water reactor spent nuclear fuel, respectively), each about 4.6 meters (15 feet) long. As with the vault approach, each sleeve would contain five spent fuel elements (i.e., in layers) within a basket or can arrangement for maintaining spacing and retrievability. Also, as with the vault approach, the number of dry storage casks would depend on the decay heat of the fuel and a cladding temperature limit [nominally, 175°C (347°F) for aluminum cladding with an air inlet temperature of 49°C (120.2°F)]. The 24-sleeve design would allow a maximum of 120 elements for fuel with 40 to 80 Watts per element of decay heat, while the 52-sleeve design would provide a minimum of 260 elements per dry storage cask with 10 to 40 Watts per element. Thus, based on the total number of elements for which the facilities are sized, 94 casks would be required, predicated on a 3-year cooldown period (i.e., less than 40 Watts per element). This value is conservative. Again, most nuclear fuel is likely to have decay heats between 10 and 40 Watts per element. Initially, fuel with higher heat loads could be unsuitable for the dry storage cask pending detailed heat transfer analysis and a final determination of limiting fuel storage temperature for aluminum-based and TRIGA-type spent nuclear fuel. However, fuel with a relatively high decay heat represents such a small percentage of currently identified spent fuel that its impact would be small, such that after 3 years of wet storage it would all be below a heat output of 40 Watts per element.

TC

Figure C-17 shows the general layout for the dry cask storage facility, predicated on a horizontal cask design.

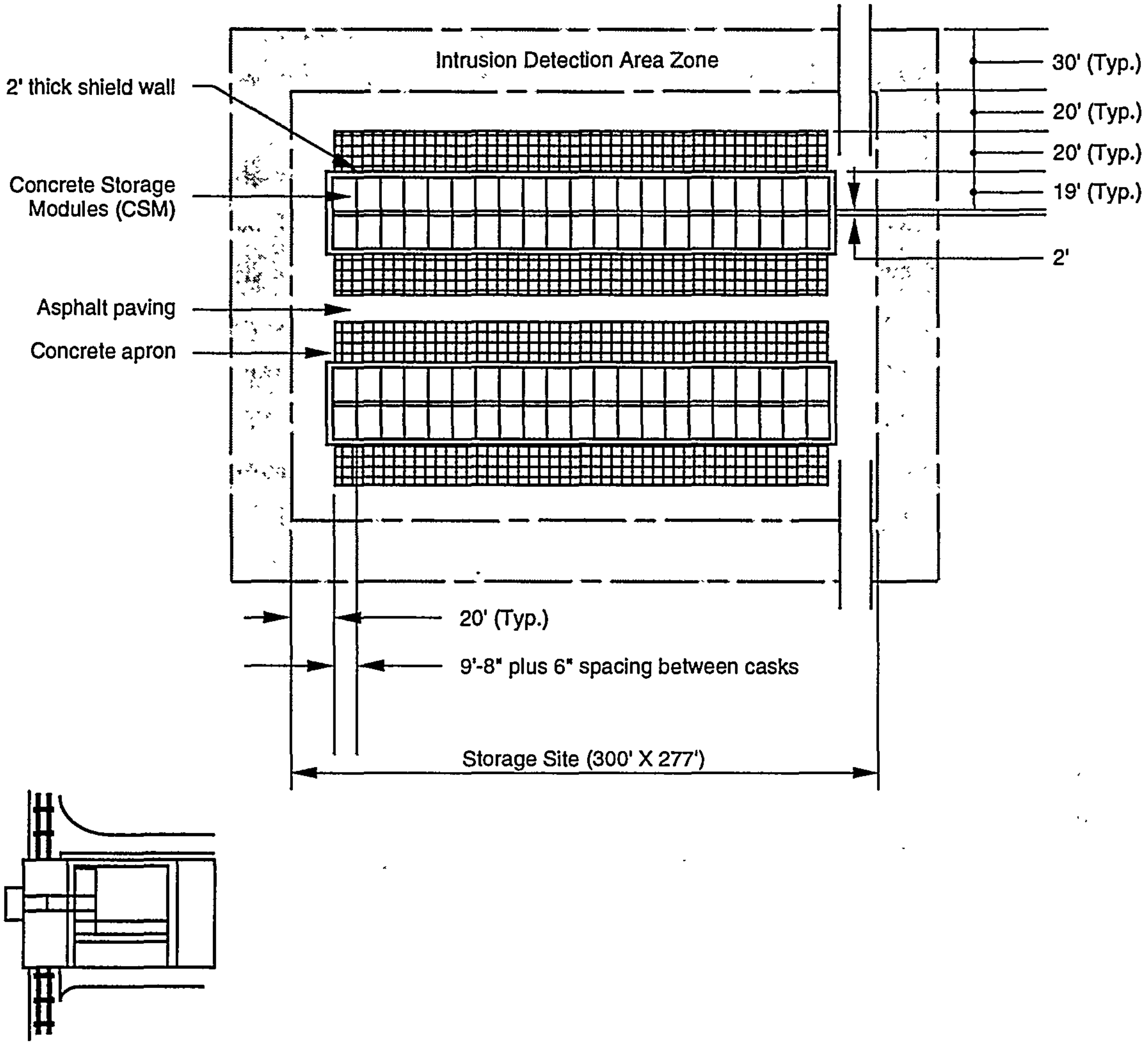
Dry storage cask technology would require a separate staging facility for fuel unloading, canning, and storage cask loading, and shipping cask maintenance. This facility would have the following operational areas:

- Transportation Cask Handling. To incorporate transportation cask maintenance, truck and railcar unloading, decontamination and washdown, radioactive material control, and cask sampling, flushing, and degassing.

- A Small Wet Storage Pool for fuel transfer and short-term storage.

- Spent Nuclear Fuel Unit Handling to provide fuel removal, decontamination, fuel drying, fuel canning, inserting with helium, and thermal measurements.

TC



10-40 W/Element Scenario
94 cask/modules with dry storage canister (up to 260 Generic Spent Fuel Elements Per Module = 25,000 storage capacity)
• Passive cooling • Retrieval requires new canister • Requires separate staging facility

Note: All dimensions shown are approximate and may only be used for a conceptual "footprint" estimate.

Figure C-17. Layout of a modular dry cask storage facility for spent nuclear fuel.

- Spent Nuclear Fuel Unit Transfer to place the fuel into the cask or canister, followed by sealing.

- Radwaste Treatment, including collection, treatment, and preparation for disposal of contaminated effluents, and radioactive waste treatment and solidification.

- Heating, Ventilating, and Air Conditioning to help ensure that contamination of workers and the environment is avoided.

The inspection and characterization facility would include a shielded dry hot cell for spent fuel analysis and examination, and for canning of leaking spent nuclear fuel. All equipment and instrumentation in the cells would be remotely operated. The facility would be maintained under negative pressure with exhaust through high-efficiency particulate air filters to mitigate the environmental effects of any radionuclide releases. It would normally be immediately adjacent to, or within, the staging facility. TC

Dry cask storage is unique because of its ability to be integrated operationally with existing facilities, which would enable faster implementation than other storage technologies. Several DOE management sites have facilities with spent nuclear fuel handling capabilities similar to the requirements of the staging facility. For dry cask storage, the spent nuclear fuel would be shipped to the existing facility and unloaded from the shipping cask. The spent nuclear fuel would be inspected, canned if it was leaking, and placed inside the storage canister. Spent nuclear fuel elements with heat loads exceeding 40 Watts per element would be stored in the existing facility to allow cooldown prior to cask storage. After filling, the canister would be sealed and placed inside the storage cask. The only new construction required would be the concrete storage pad (for vertical casks) or concrete storage modules (for horizontal casks).

C.14.3 ACTINIDE PACKAGING AND STORAGE FACILITY

The Actinide Packaging and Storage Facility would provide stabilization, repackaging, and storage of special nuclear material (SNM) to meet the requirements of DOE-STD-3013-94, *Criteria for Storage of Plutonium Metals and Oxides* (DOE 1994h). The criteria presented in this standard are designed to help ensure the safe storage of the material for long periods of time (e.g., 50 years). The material would be packaged in at least two corrosion-resistant containers (a container within a container). The packaging process would not use plastics, hydrogenous compounds, or organic material due to the radiolytic effects (principally gas generation) of long-term plutonium storage in the presence of these L11-6 materials. The outer container would be welded shut in an inert atmosphere to ensure weld joint and container material integrity. The outer container would have to meet the packaging requirements of 10 CFR Part 71 and to withstand the worst-case anticipated pressurization scenario, which assumes 100 percent radiolysis of retained moisture, helium generation for 50 years, and temperatures as high as 204°C (400°F).

DOE has identified actions that could accelerate the construction and startup of the Actinide Packaging and Storage Facility. The normal project management process for a facility that will be a major systems acquisition (i.e., more than $100 million) has seven distinct steps, including four "key decision" points, ranging from the initial steps of identifying facility functional design and performance criteria to the final step of approving the facility for operation. DOE plans to consolidate several of the initial project steps and to integrate and accelerate design and construction work through the use of design and construction proposal packages. DOE estimates these actions could reduce design and construction time such that the facility could be ready for operation in about 6 years.

Option 1 - New Facility

L11-6 The new facility would have a total area of approximately 3,700-square meters (40,000-square feet). It would be north of Building 235-F and east of Building 247-F in F-Area. (See Figure C-18.) It would consist of a 2,100-square meter (23,000-square foot) hardened structure containing the Material Access Area (MAA). The MAA would be bounded on the south and east by a contiguous 1,500-square meter (16,000-square foot) "soft" structure containing offices, heating, ventilation, and air conditioning support equipment, and other administrative functions.

The facility would consist of process areas and equipment for truck unloading and loading, material confirmation, shipping, package unpackaging and packaging, accountability measurements, safety evaluation International Atomic Energy Agency inspections, repackaging, waste management, a vault room, and full support and administration functions. Utilities and services would include electricity, potable process water, chemical storage, steam, compressed air, standard and high-efficiency particulate air (HEPA) filtered ventilation, and communications. About 130 persons would operate this facility.

DOE has identified two methods to create this facility. The first would be to construct a new facility within the F-Area industrial complex. This area has been disturbed repeatedly over the last 40 years by the addition of various facilities. The second method would be to modify Building 235-F. Figure C-18 shows both of the locations of 235-F and the new facility.

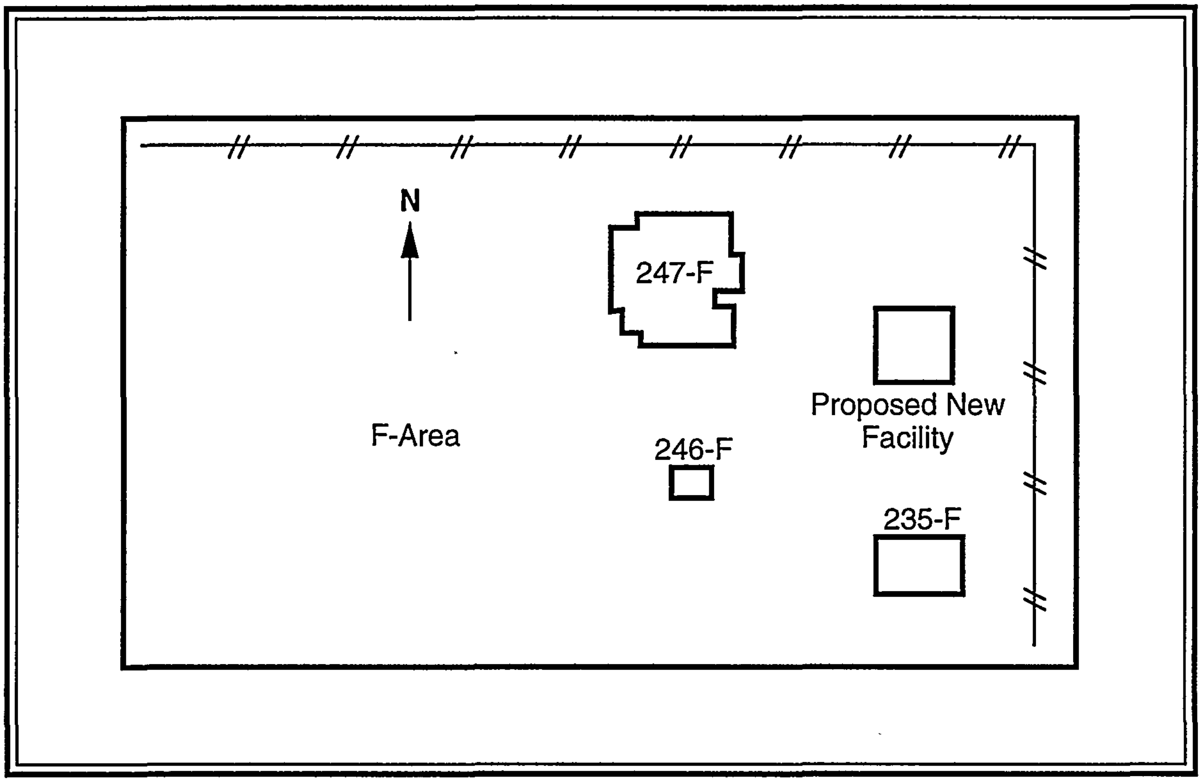


Figure C-18. Proposed location of Actinide Packaging and Storage Facility. L11-6

After the material had been packaged in accordance with the requirements of the DOE standard, it would be stored in a heavily shielded concrete vault. This vault (and the portion of the facility used for packaging the material) would meet the requirements for plutonium-handling facilities established in DOE Order 6430.1A. The facility would have appropriate fire detection and suppression systems, a nuclear-grade ventilation system, radiation and contamination detection systems, material confinement systems, waste management systems, and safeguards and security systems.

For vault surveillance and container handling an Automated Guided Vehicle (AGV) and automation will be used to minimize exposure. Additionally, a bagless transfer operation will be used to minimize exposure for the transfer of SNM from the existing containers to containers designed for long-term storage.

The exhaust stack would have continuous isokinetic sampling and monitoring of radionuclides. A waste management area would be used to sort, compact, and monitor solid waste. Ventilation air for unhardened areas that do not contain special nuclear materials would be exhausted to the atmosphere without filtration because there would be no risk of airborne radioactivity. Ventilation for the staging and storage building would be once-through with two stages of DOP-testable HEPA filtration on the exhaust. This exhaust stream would not affect the environment. CO_2 (dry ice) pellet blasting would predominant the decontamination method used in the facility. Its use would help to minimize waste generation, including liquids. All waste leaving the Material Access Area in the facility would be monitored.

The Waste Management area would process contaminated or potentially contaminated waste generated during normal storage facility operations. All other waste materials would be handled by the waste generator (shops, offices, etc.). The small volume of waste generated during storage operations would be collected in each facility area and taken to a central waste management area for screening, measurement, assaying, and preparation for shipment from the facility. The waste materials would be sorted, categorized, and transferred to the appropriate SRS waste facility for disposal. Because there are extensive waste processing facilities at the SRS, there would be minimal waste processing in the storage facility.

Option 2 - Minimum Upgrade of Building 235-F

L11-6

Building 235-F is a two-level hardened structure with 2,200 square meters (24,000 square feet) on each level. At present, approximately 50 percent of this area is available for other uses, assuming some decontamination and restoration of contaminated areas and the modification of office spaces, change rooms, and some maintenance and support areas.

The Building 235-F minimum upgrade project would provide a facility and equipment for truck unloading, material confirmation, shipping package unpackaging, repackaging, accountability measurement, safety evaluations, adequate vault storage space (1925 storage positions), and International Atomic Energy Agency inspections. The capability for waste management is available in the building or will be provided by other projects. Fire protection, emergency power, and life safety systems upgrades would be only for those areas of 235-F to be modified.

Utilities and services would include electricity, potable and process water, chemical storage, steam, compressed air, standard and high-efficiency particulate air filtered ventilation, and communications. About 130 persons would operate the 235-F facility after the upgrade (180 during repackaging operations).

An AGV would be used for surveillance and container handling to minimize exposure. In addition, a bagless transfer operation and digital radiography would be used to minimize exposures for the repackaging of special nuclear material from the existing containers to containers designed for long-term storage. The processes and equipment must be placed where room is available in Building 235-F. Therefore, worker exposure is highest in this option because the ideal equipment arrangement and material flow is not possible.

The exhaust stack would have continuous isokinetic sampling and monitoring of radionuclides. A waste management area will be provided to sort, compact, and monitor solid waste.

A new nuclear-grade HVAC system would be part of the project. CO_2 (dry ice) pellet blasting would be the predominant decontamination method used in the facility. Its use would help to minimize waste generation, including liquids. All waste leaving the Material Access Area in the facility would be monitored. TE

The Waste Management area would process contaminated or potentially contaminated waste generated during normal storage facility operations. All other waste materials would be handled by the waste generator (shops, offices, etc.). The small volume of waste generated during storage operations will be collected in each facility area and taken to a central waste management area for screening measurement, assaying, and preparation for shipment from the facility. The waste materials will be sorted, categorized, and transferred to the appropriate SRS waste facility for disposal. Because there are extensive waste processing facilities at SRS, there would be minimal waste processing in Building 235-F.

C.14.4 FB-LINE MODIFICATION

The FB-Line modifications would provide the capability to assay containers of actinides (such as plutonium), to heat plutonium oxide, and to package plutonium oxide and metal to meet the DOE storage criteria (DOE 1994h). TE

An existing area of FB-Line would be used for the installation of the required assaying equipment. This equipment would consist primarily of digital radiography screening equipment that could be put in place with minor (if any) modifications.

The heating capability would be provided through modifications to furnaces installed in FB-Line. Packaging would be enabled by connecting a new glovebox to the existing FB-Line mechanical line.

An existing glovebox would be removed and replaced with a larger version that would include equipment to load storage containers and seal them in an inert atmosphere (e.g., helium). It also would provide the capability to remove loaded containers via a sphincter seal port, which would prevent the need for plastic wrapping.

C.14.5 F-CANYON VITRIFICATION FACILITY

TC After evaluating the time required to construct a vitrification facility, DOE determined the optimum case would be to modify a portion of the F-Canyon. This facility would provide the support structure, utilities, and services necessary for vitrification and would require only the installation of new equipment. DOE estimates the time required to install the vitrification equipment and make it ready for operation would be about 3-1/2 years. This time would be minimized by using knowledge gained from Defense Waste Processing Facility vitrification development and testing, by accomplishing as much design and development work as possible in parallel with National Environmental Policy Act evaluations, and by using standardized equipment as much as possible.

The proposed F-Canyon Vitrification Facility would vitrify actinides such as americium, curium, and plutonium. It would provide shielding, remote handling and viewing capability, process area ventilation, and removable rack and module type construction to enable the installation of completely tested process modules. The facility, originally designed to process californium-252, consists of:

- Eight shielded hot cells; six shielded process rack positions; two shielded analytical [1.5 meters (5 feet) of concrete/1.5 meters (5 feet) of leaded glass] areas
- Computer room and cation column stream monitor room
- Column process pump and general instrument room
- Cold feed makeup, storage, and delivery tankage
- Rack hot and cold water systems
- Small equipment entry sphincter
- Canyon equipment such as a feed tank, waste tank, feed evaporator, associated jumpers, samplers, etc.

- The hot canyon crane, which would be used to remove solid waste and product packages and other necessary work by removing the canyon cell covers over the rack area

The facility would be modified for vitrification by removing equipment such as racks and installing new equipment modules in several cells. The new equipment would include solution and powdered glass feed systems, a melter, an off-gas system, a glass canister loading and cap closure system, and a materials handling system. In addition, the existing in-cell crane, master-slave manipulators, transfer equipment, services, and utilities would be refurbished as necessary. Geometrically favorable equipment would be installed in F-Canyon for the concentration of plutonium solutions.

The following list describes the proposed modifications to F-Canyon:

Process Racks

- Removal of 2 to 3 existing process racks

- Fabrication of storage boxes for the existing process racks

- Equipment development, design, procurement, and fabrication of two new process modules. The new modules would include the following:

  - All necessary tankage
  - The melter
  - Melter offgas system
  - Glass canister handling equipment
  - Individual sumps for each module

TC

Analytical Racks

- Renovation as required to support in-cell sample analysis and sample transfer to the second level shielded transfer facility.

In-Cell Crane

- Removal and refurbishment of the in-cell crane and controls
- Installation and checkout of the in-cell crane

Manipulators

- Checkout of existing manipulators
- Refurbishment as required, such as cable replacement or boot replacement
- Replacement of manipulators as required

Services and Utilities

TC

- Checkout and renovation as required for the cooling water system
- New steam and condensate services
- Checkout and renovation or modification of cold feed facilities
- Reactivation of canyon deluge valve fire system or renovation of the Halon system
- Electrical upgrades for melter power supply
- Installation of new sample facilities in the Hot Sample Aisle
- Replacement of in-cell HEPA filters

The following process descriptions for the vitrification of americium and curium solutions and plutonium solutions are based on the use of remotely operated equipment in the F-Canyon and vitrification process knowledge. The process would be accomplished by adjusting or preparing the solutions for vitrification and then vitrifying them.

- **Feed Preparation.** Americium and curium solutions would be adjusted and processed to separate actinides (by precipitation) from other metals (primarily iron). This step would reduce the number of glass cylinders produced by eliminating unnecessary metals from the feed stream and concentrating the americium and curium. Plutonium solutions would be evaporated to reduce volume and to achieve greater processing efficiency.

- **Vitrification.** The solutions of concentrated actinides would be fed to the F-Canyon Vitrification Facility melter where they would be flash-evaporated and the americium and curium nitrates would be oxidized to form an americium and curium oxide. The oxide would be combined with molten glass. The melt would flow into stainless-steel canisters where it would solidify. The cooled canisters would be sealed, decontaminated, and overpacked for storage.

C.14.6 LOW ENRICHED URANIUM OXIDE WAREHOUSE

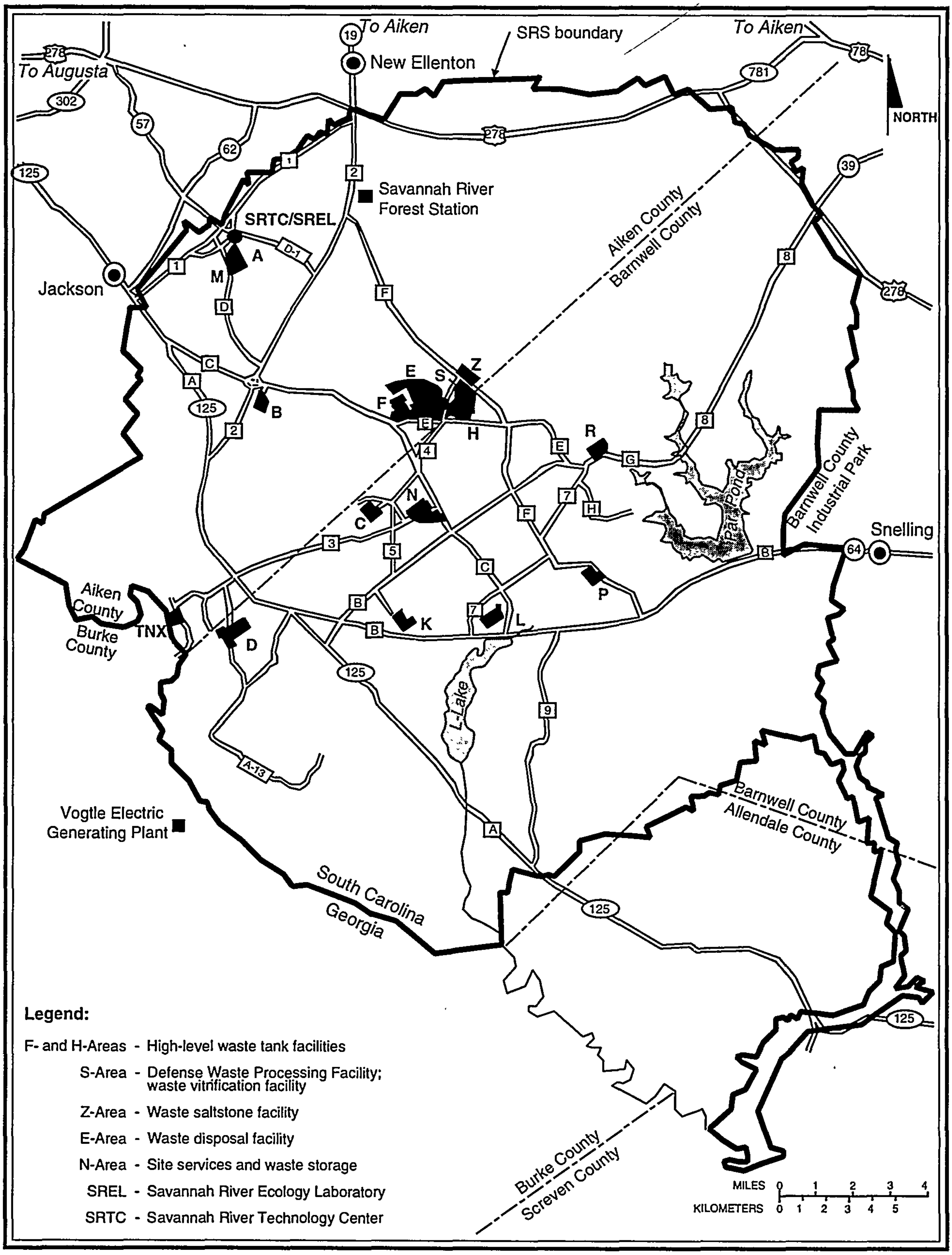
This facility would have an area of approximately 560 square meters (6,000 square feet) and would be of standard construction (i.e., concrete slab with a metal building). It would have an administrative area and the remainder would be storage space. The facility would be supplied with normal industrial utilities and services (potable water, sewer, electricity, compressed air, steam, ventilation, etc.). Appropriate security would be provided. The facility would be in the existing F- or H-Area on previously disturbed land. TE

C.15 Waste Management Facilities

This section describes the waste management facilities associated with wastes generated by the alternatives evaluated in this EIS; these include high-level waste (the borosilicate glass produced in the Defense Waste Processing Facility and the saltstone), low-level solid wastes, transuranic wastes, and hazardous and mixed wastes. These wastes, their associated facilities, and their management are addressed in the Supplemental Environmental Impact Statement for the Defense Waste Processing Facility (DOE 1994g) and the Savannah River Site Waste Management Draft Environmental Impact Statement (DOE 1995c). TE This section describes these facilities to improve understanding of the consequences of the stabilization actions; it also describes the facilities associated with surface-water releases discussed in Section 4.5.

DOE treats and stores wastes generated from onsite operations in waste management facilities, most of which are in E-, F-, H-, N-, S-, and Z-Areas (Figure C-19). TE Major facilities include the high-level waste tank farms, the Low-Level Radioactive Waste Disposal Facility; the F- and H-Area Effluent Treatment Facility; the Defense Waste Processing Facility (undergoing startup testing); and the Consolidated Incineration Facility (under construction).

DOE stores liquid and solid wastes at the Site. Liquid high-level radioactive waste is stored in underground storage tanks (Figures C-20 and C-21), which are managed in accordance with Federal laws, South Carolina Department of Health and Environmental Control (SCDHEC) regulations, and DOE Orders. Figure C-22 is a flow diagram for liquid high-level radioactive waste handling from the point of generation to conversion to borosilicate glass and saltstone. TE



TE | **Figure C-19.** Savannah River Site, showing waste management facilities.

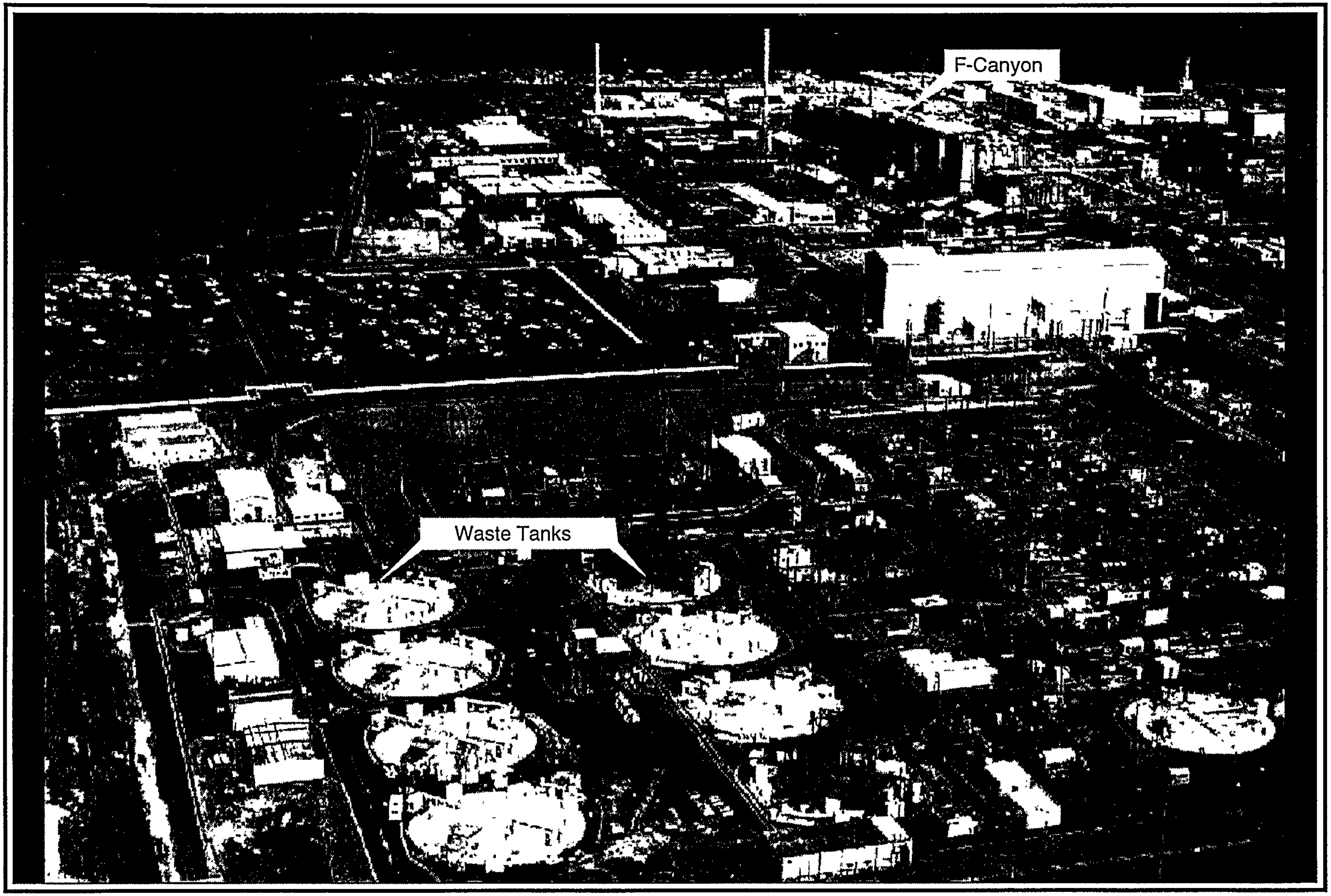


Figure C-20. F-Area tank farm (view toward the north, with 9 of the 22 F-Area liquid high-level waste tanks).

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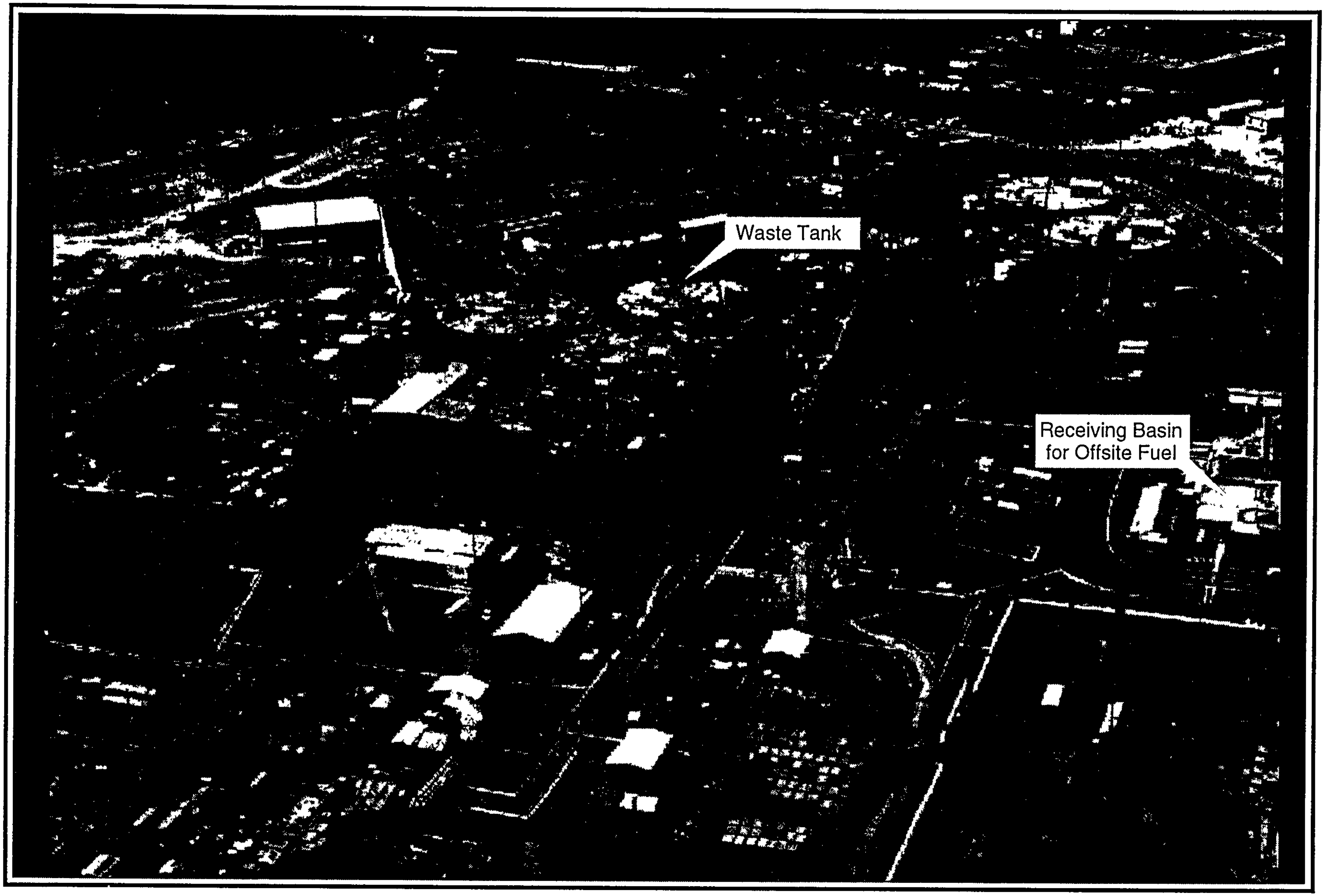

Figure C-21. H-Area tank farm (view toward the west, showing 5 of the 29 H-Area liquid high-level waste tanks.)

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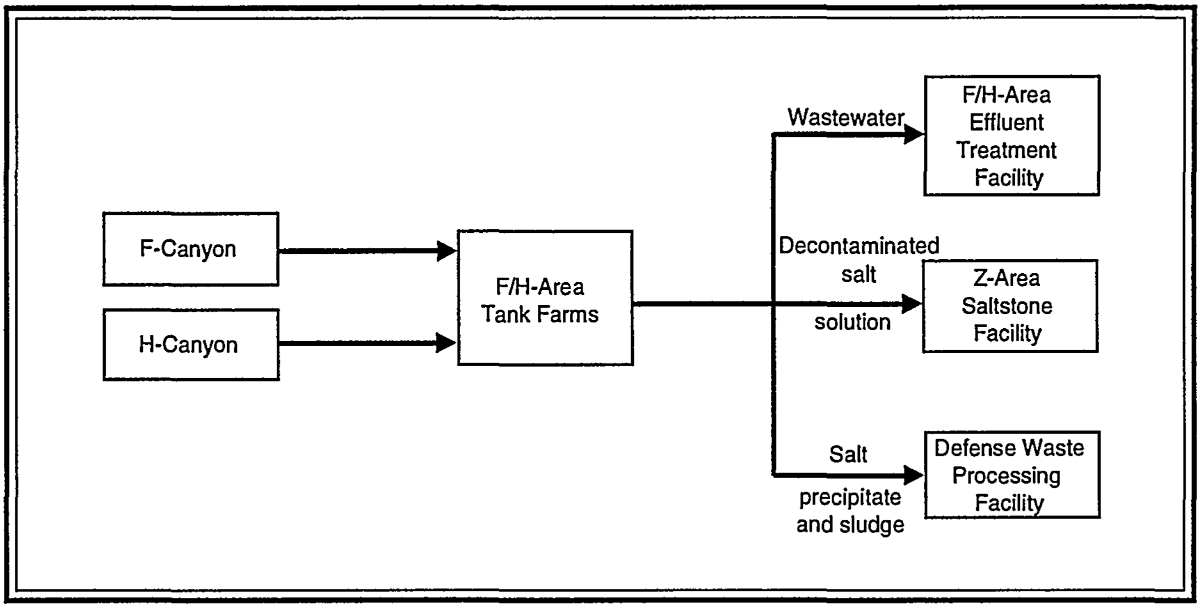

Figure C-22. Liquid high-level waste handling. TE

Low-level solid wastes (LLW) contain low radioactive levels and are not classified as transuranic waste, spent nuclear fuel, or byproduct material. Low-level solid wastes have been generated by all the nuclear facilities at the SRS, and they would be generated by the stabilization alternatives described in this EIS. The generating facility packages and monitors the wastes and sends it to the Low-Level Radioactive Disposal Facility in E-Area (Figure C-23). These wastes are disposed of in accordance with South Carolina Department of Health and Environmental Control requirements and DOE Orders. TE

DOE stores solid transuranic (TRU) wastes on an interim-status on storage pads in accordance with SCDHEC requirements and DOE Orders. These wastes are generated, packaged, and monitored primarily in the F-Area and H-Area B-Lines and transferred to E-Area for storage. Figure C-24 shows surface storage of the transuranic waste in concrete culverts. TE

Hazardous waste is waste that exhibits hazardous characteristics (ignitability, corrosivity, reactivity, or toxicity) as defined by the U.S. Environmental Protection Agency (EPA) or SCDHEC. Hazardous waste includes organic liquid, debris, or sludges; aqueous liquid, debris, or sludges; metal debris; glass debris; inorganic sludges; and soils. Mixed waste is hazardous waste that contains radioactivity; this type of waste would be generated in the stabilization alternatives.

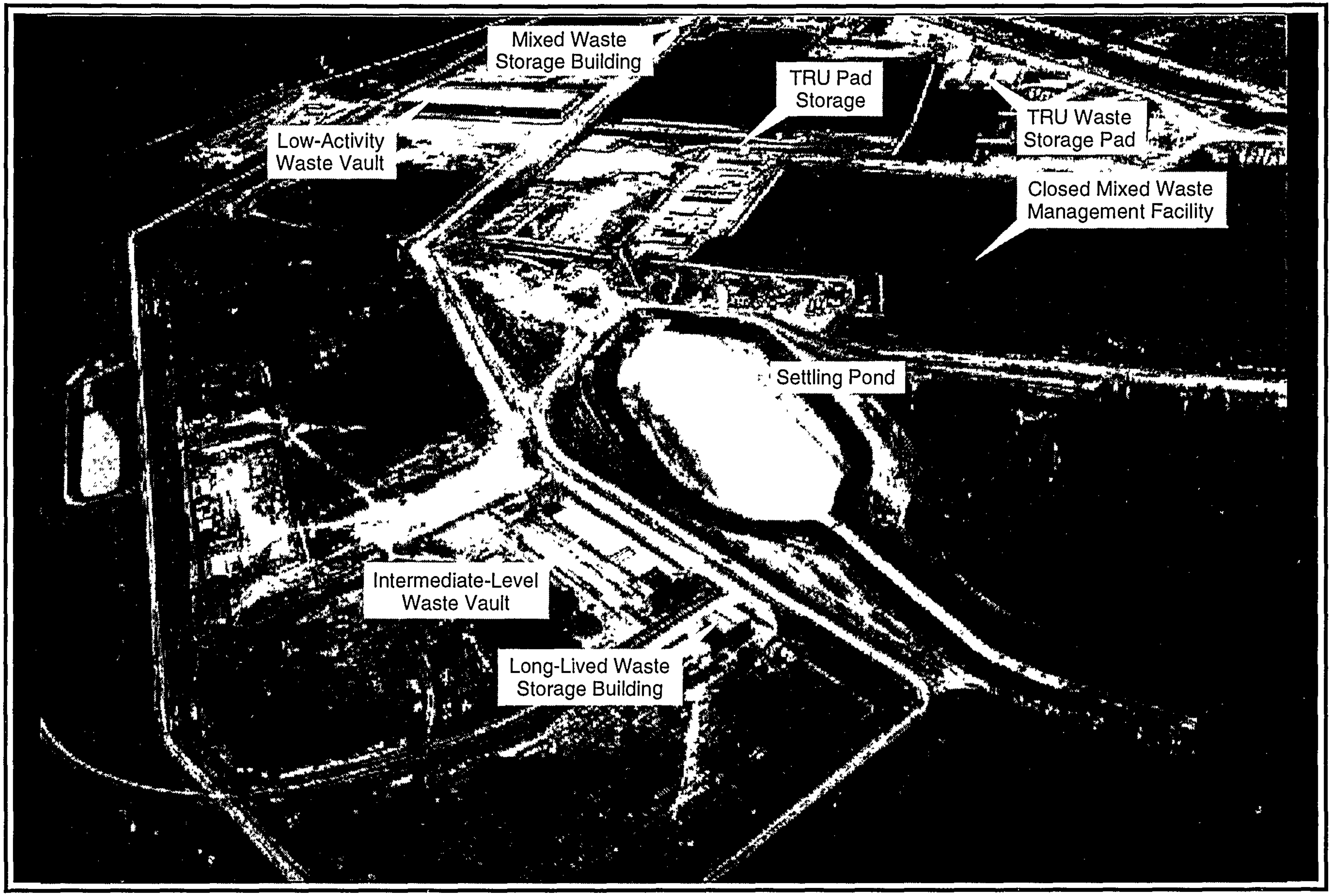

Figure C-23. Low-Level Radioactive Waste Disposal Facility (E-Area).

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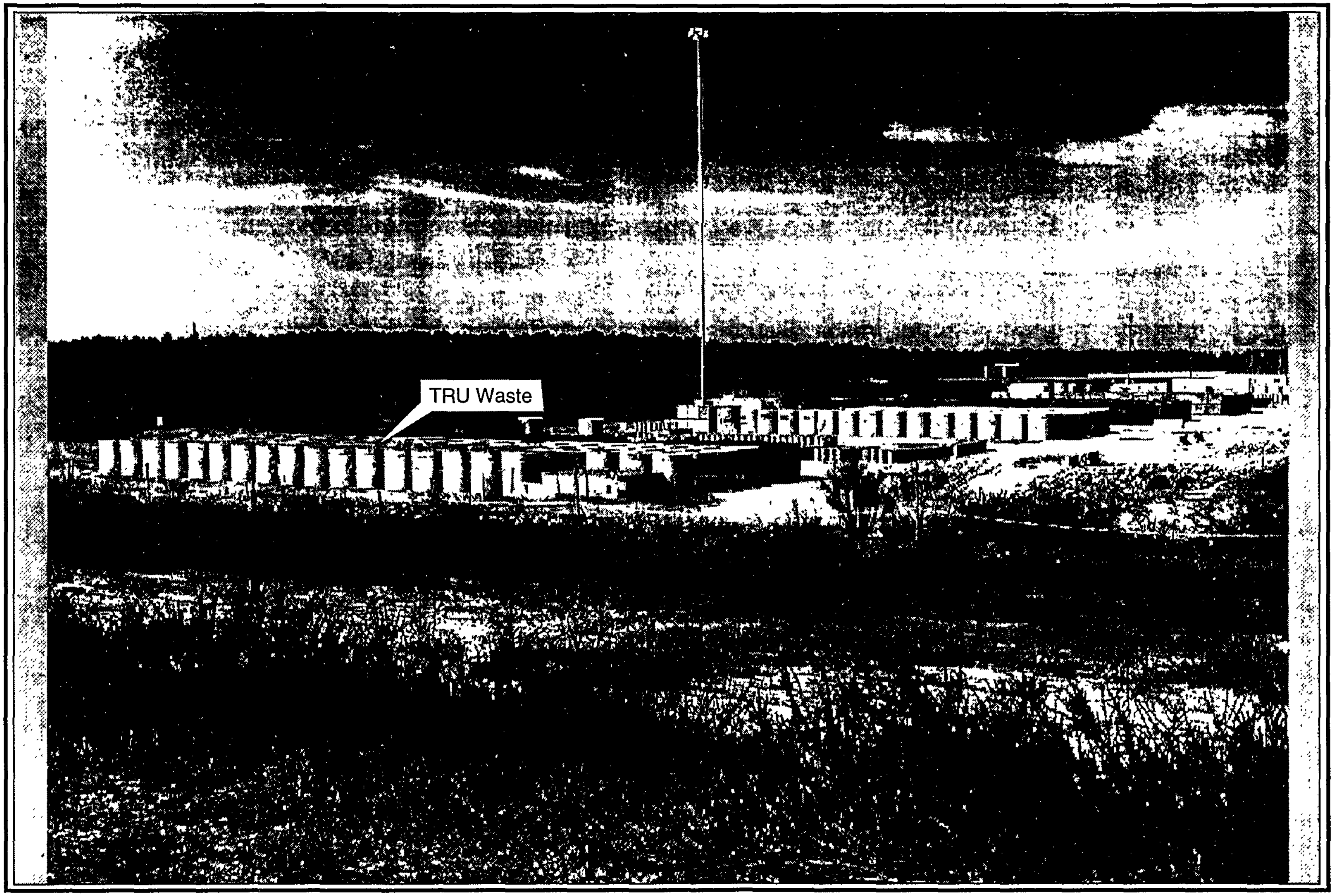

Figure C-24. Transuranic waste storage pads (E-Area).

TE

SRS mixed wastes are stored in permitted or interim-status facilities such as the hazardous waste storage facilities (building and pads) and in the mixed waste storage buildings (Figure C-25 and C-26, respectively). Figure C-27 shows waste handling for other forms of waste at the Site.

Wastewater contaminated with low-level radioactivity is treated at the F- and H-Area Effluent Treatment Facility (ETF), an SCDHEC-permitted Industrial Wastewater facility (Figure C-28). The wastewater feed to the ETF is primarily from the F- and H-Area Outside Facilities described in Section C.6; the ETF effluent is the major source of radionuclides released to SRS streams for the alternatives discussed in this EIS (see Section 4.5).

C.15.1 LIQUID HIGH-LEVEL RADIOACTIVE WASTE

This section summarizes the facilities and processes used to produce the borosilicate glass and saltstone from the high-level waste that would be generated in the nuclear material stabilization alternatives. The Supplemental EIS on the Defense Waste Processing Facility (DOE 1994g) contains more detail. The liquid high-level radioactive waste is highly radioactive material resulting from the reprocessing of spent nuclear fuel; such liquid waste would be produced directly in the stabilization of nuclear materials, primarily in F- and H-Canyons.

This waste is alkaline; that is, it has been neutralized before discharge to the underground carbon steel waste tank (see Figures C-19 and C-20). It is stored until short-lived radionuclides decay to acceptable levels and insoluble components of the waste (about 5 to 10 percent) have settled out to form a sludge layer on the tank bottom. The liquid waste is evaporated to reduce its original volume and to immobilize it as crystallized salt. The Supplemental EIS for the Defense Waste Processing Facility (DOE 1994g) provides details on this process. Condensed evaporator overheads are transferred to the F- and H-Area Effluent Treatment Facility, which decontaminates routine process effluents from F- and H-Areas. The salt fraction will be processed further via in-tank precipitation to separate it into a highly radioactive portion for vitrification at the Defense Waste Processing Facility and a low radioactive salt solution stream for stabilization as saltstone and disposal at the Z-Area Saltstone Facility. This stabilization process will include the blending of liquid salt solution with cement, slag, and flyash to produce the saltstone grout for the concrete disposal vault where the saltstone becomes solidified concrete waste. The sludge will be washed to remove impurities and then vitrified in the Defense Waste Processing Facility. The vitrified product, a molten borosilicate glass, will be poured into a stainless-steel canister that, when full, will be sealed with a temporary plug. Typical glass material will be approximately 72 percent glass frit and 28 percent waste. Each canister

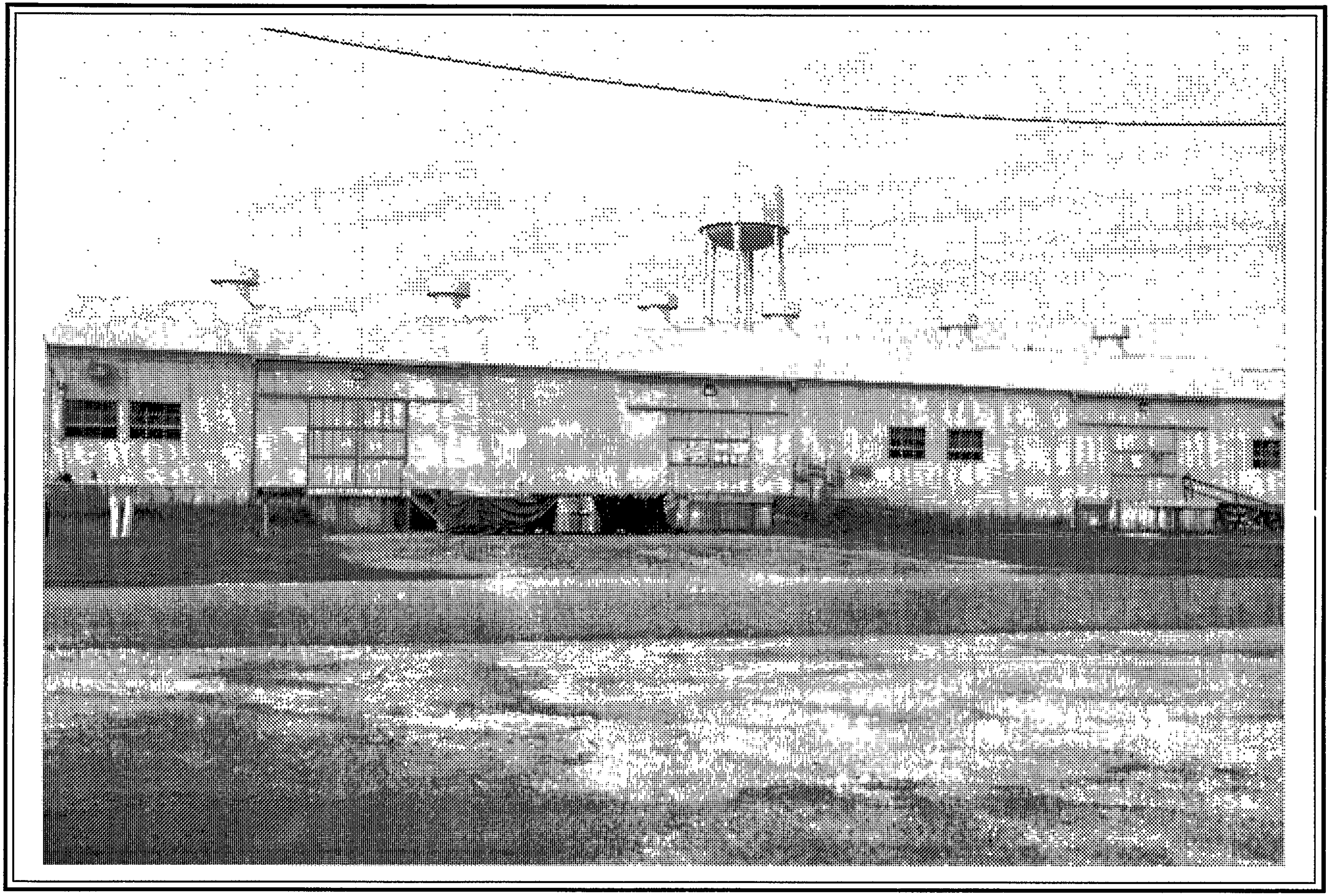

Figure C-25. Hazardous waste storage facility (B-Area).

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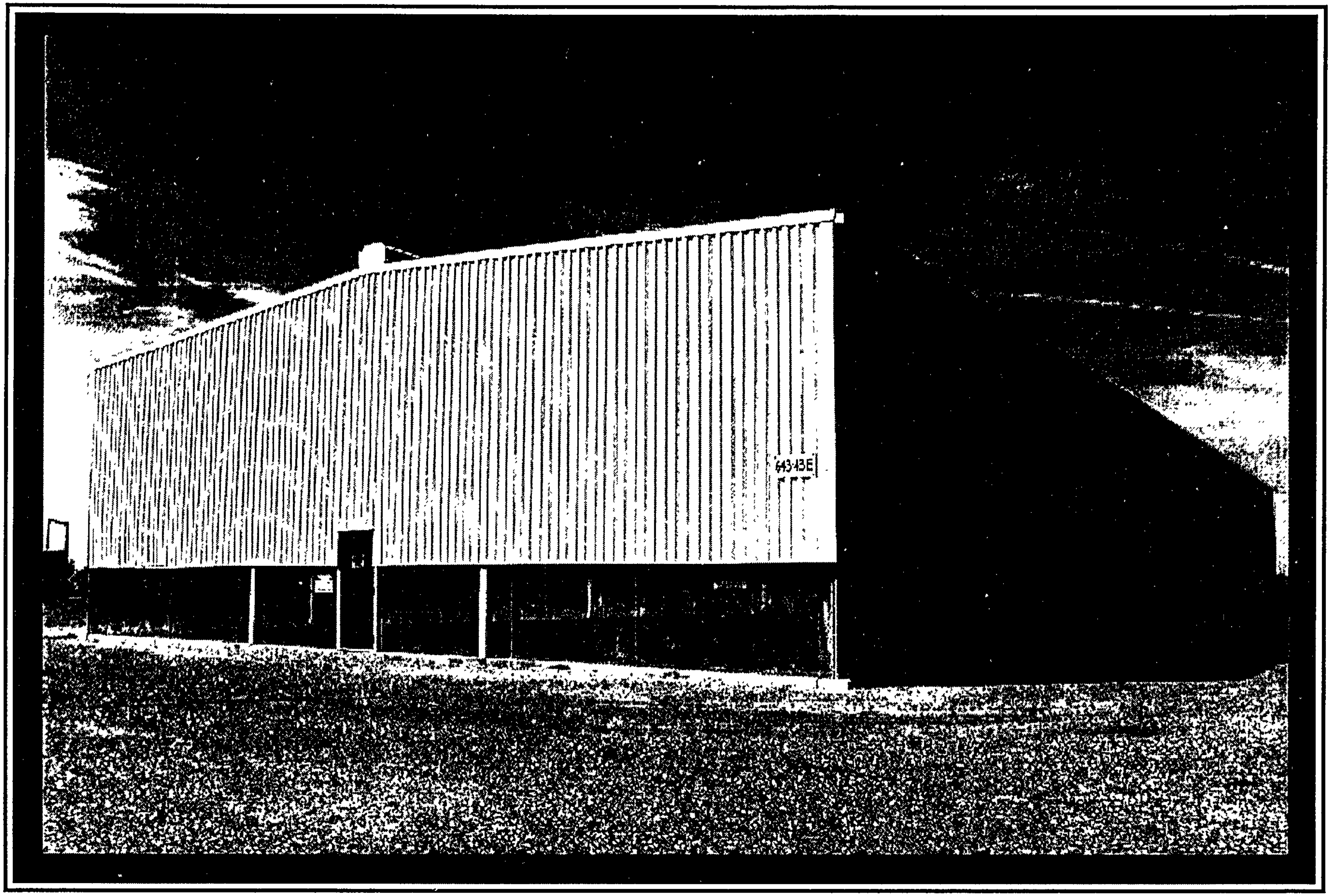

Figure C-26. Mixed waste storage building (E-Area).

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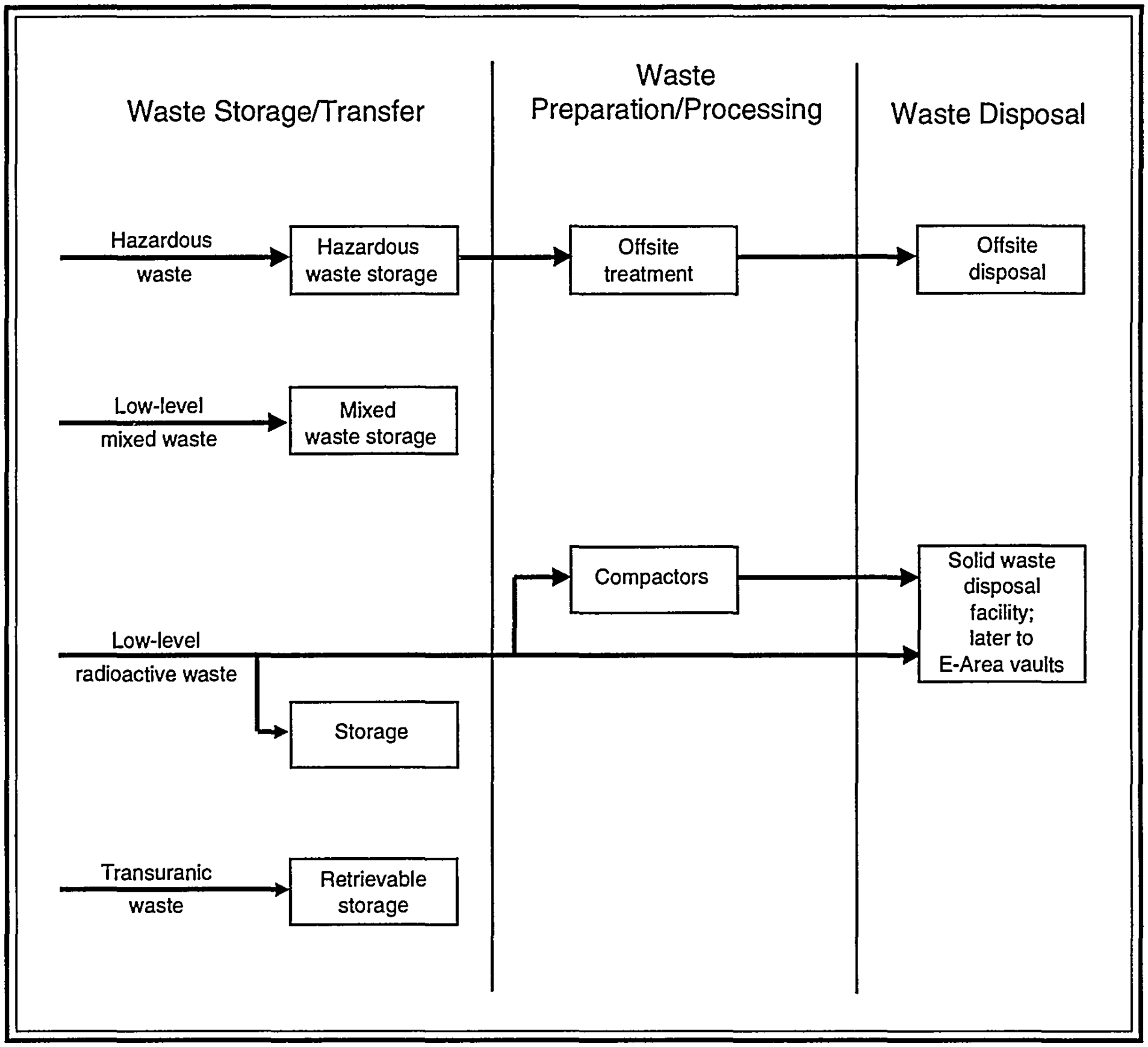

Figure C-27. SRS waste handling processes. | TE

will contain approximately 1,680 kilograms (3,700 pounds) of glass. After decontamination and the final welded closure, the canister will be transferred to the Glass Waste Storage Building.

C.15.2 LOW-LEVEL SOLID WASTE

As indicated above low-level waste contains low levels of radioactivity and is not classified as transuranic waste, spent nuclear fuel, or by-product materials. At the SRS, low-level waste is further categorized for onsite disposal in the Low-Level Radioactive Waste Disposal Facility according to its waste category and its surface radiation dose. The primary categories include low-level, intermediate-level, and long-lived wastes. The SRS also distinguishes between wastes that have low surface

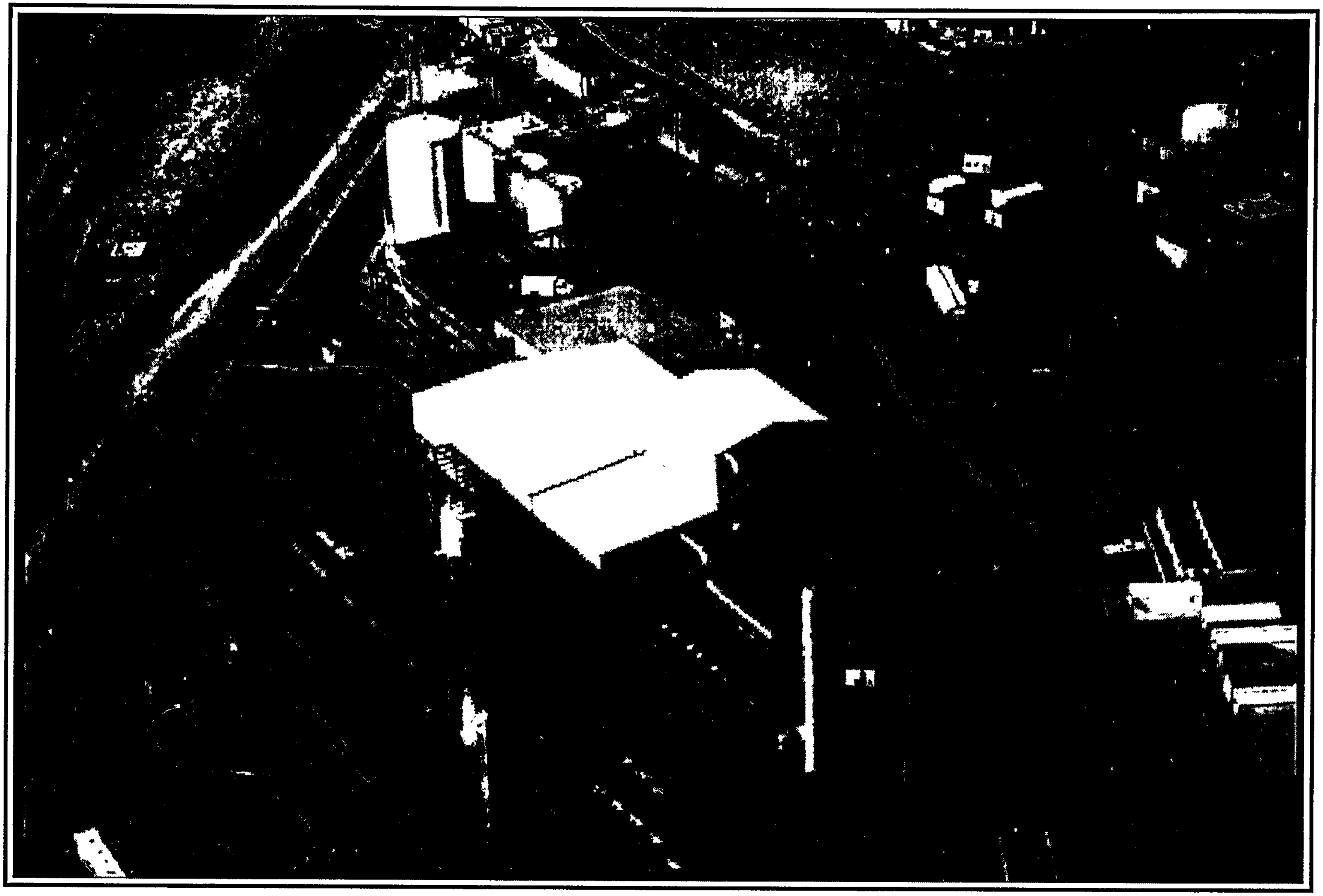

Figure C-28. F/H-Area Effluent Treatment Facility (H-Area).

TE

radiation doses and can be handled directly and those that require remote handling. Another type of low-level waste contains significant alpha activity (between 10 and 100 nanocuries per gram), which is managed as if it were transuranic waste. This waste is discussed in Section C.15.3.

Low-activity waste is loaded in steel boxes and can be shipped either to the Low-Level Radioactive Waste Disposal Facility in E-Area or to the H-Area compactor for compaction. If the waste is compacted, more waste is added to the steel boxes and compacted until the boxes are full. The boxes are taken to the Low-Level Radioactive Disposal Facility in E-Area (see Figure C-23) where they are placed in engineered low-level trenches. The trenches are several acres by 6 meters (22 feet) deep, with sloped sides and floors directing drainage to a collection sump. When a trench is full of boxes, DOE backfills and covers it with at least 1.8 meters (6 feet) of soil. TE

DOE packages intermediate-level wastes according to the waste form and disposes of them in slit trenches. Some intermediate-level waste, such as contaminated equipment components, is wrapped in canvas before disposal.

DOE stores long-lived wastes, such as resins, in temporary facilities until the long-lived waste storage building in E-Area can begin operations. This building (Figure C-29) will provide storage until DOE develops treatment and disposal technologies. TE

To ensure improved containment the SRS has developed a new disposal facility known as the E-Area vaults, which began receiving low-level radioactive waste in November 1994. This facility ultimately will include low-activity, intermediate-level nontritium, and tritium vaults (Figures C-30 and C-31). TE

C.15.3 TRANSURANIC SOLID WASTE

Transuranic solid waste is waste contaminated with alpha-emitting radionuclides that have a greater atomic weight than uranium (92), half-lives greater than 20 years, and concentrations greater than 100 nanocuries per gram of waste. The buried and stored wastes, which contain concentrations of transuranic radionuclides between 10 and 100 nanocuries per gram, are referred to as alpha-contaminated low-level waste. Such waste is managed like transuranic waste because its physical and chemical characteristics are similar and because similar waste examination processes will be used to determine its final disposition. The SRS stores waste containing 10 to 100 nanocuries of alpha activity per gram with transuranic wastes until DOE can determine its disposal requirements. At present, there are no treatment facilities or disposal capacities for transuranic waste; however, DOE intends to retrieve, repackage, certify, and ship all transuranic wastes off the Site for final disposition.

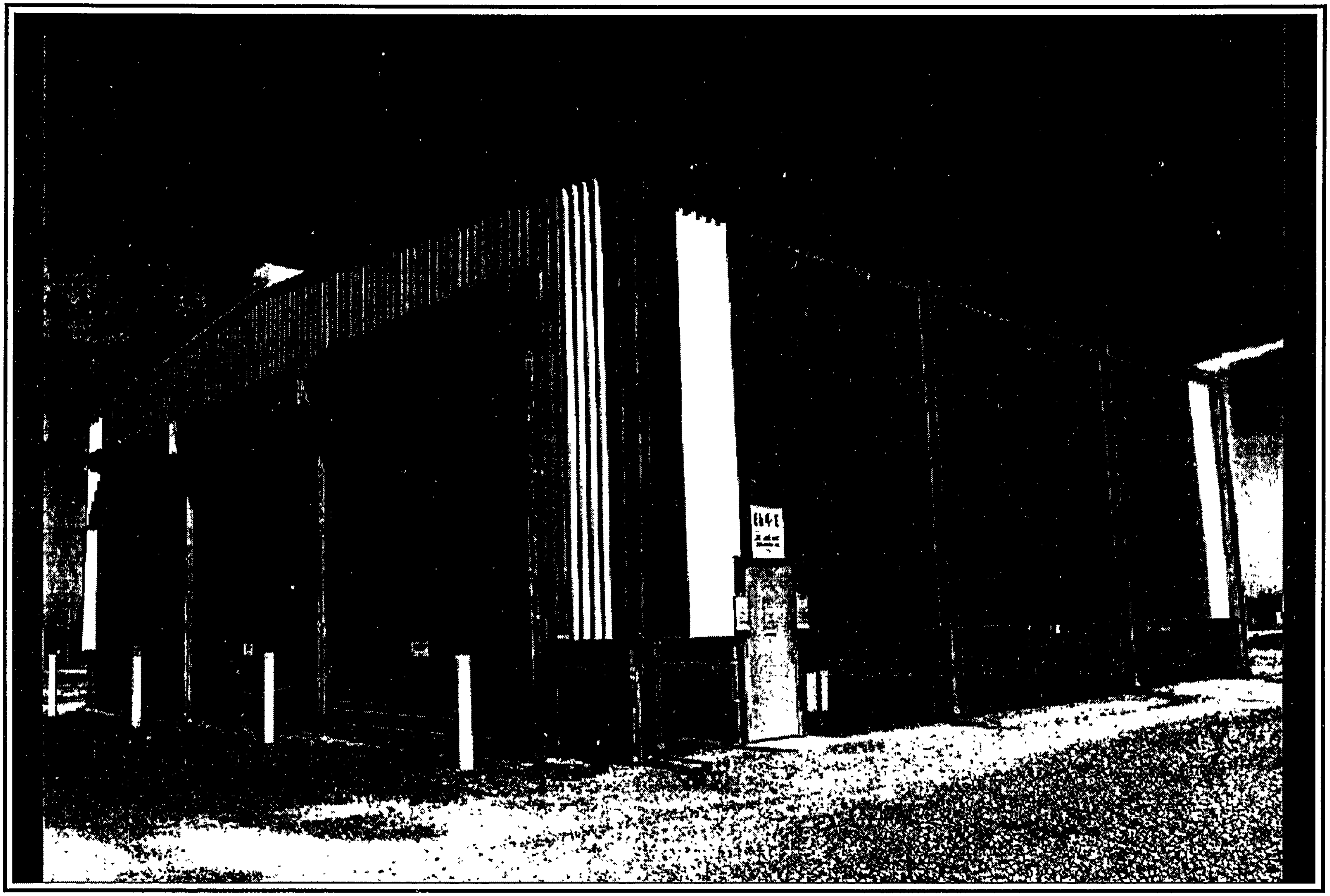

Figure C-29. Long-lived waste storage building (E-Area).

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Figure C-30. Low-activity waste vault (E-Area).

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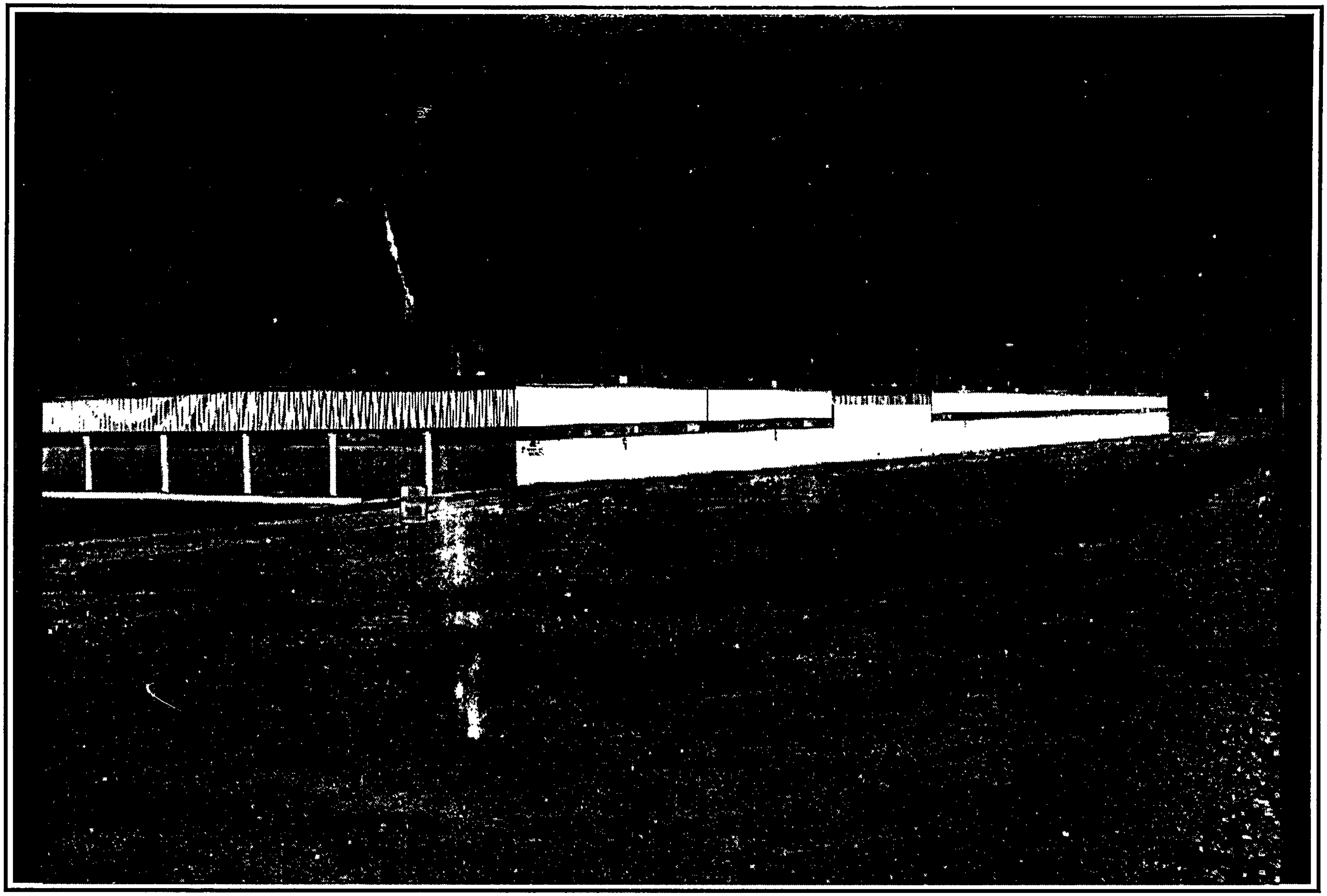

Figure C-31. Intermediate-level nontritium and tritium waste vaults (E-Area).

TE

Historically, DOE used three types of retrievable storage for transuranic waste at the SRS. Transuranic waste generated before 1974 is buried in approximately 120 below-grade concrete culverts in the Low-Level Radioactive Waste Disposal Facility. Transuranic waste generated between 1974 and 1986 is stored on five concrete pads and one asphalt pad that have been covered with approximately 1.2 meters (4 feet) of native soil. DOE stores waste generated since 1986 on 13 concrete pads that are not covered with soil. SRS transuranic waste includes mixed transuranic waste stored on Pads 1 through 17, which operate under interim status approved by the South Carolina Department of Health and Environmental Control (Figure C-32). DOE uses Pads 18 and 19 to manage nonhazardous transuranic wastes only. DOE filed a Resource Conservation and Recovery Act (RCRA) Part A permit application to describe the waste and facilities for additional storage of transuranic mixed waste on Pads 20 through 22, which are currently empty. These pads are in the Low-Level Radioactive Waste Disposal Facility. TE

C.15.4 HAZARDOUS AND MIXED WASTE

Hazardous waste is defined as discarded materials (both liquid and solid) that are either characteristically hazardous or are listed as hazardous under the Resource Conservation and Recovery Act. Characteristically, hazardous materials are corrosive, ignitable, reactive, or toxic. Wastes listed as hazardous include chemical makeup wastes, unused solvents, and discarded commercial chemicals that do not contain radionuclides. If they are contaminated with radionuclides, they are separated as mixed waste.

Mixed waste contains both hazardous waste (subject to the Resource Conservation and Recovery Act) and source, special nuclear, or byproduct material (subject to the Atomic Energy Act of 1954). Mixed waste is further classified according to its radioactive component. Low-level mixed waste is managed with its hazardous components as its primary consideration, while high-level and transuranic mixed wastes are managed with their radioactive component as the primary consideration.

The SRS F- and H-Canyons and B-Lines generate very little hazardous waste; most is mixed waste, which includes process wastes and solvents.

At the SRS, hazardous waste is stored temporarily at hazardous waste storage facilities (like that shown in Figure C-25) in buildings in B- and N-Areas and on adjacent SRS storage pads before shipment to offsite permitted treatment, storage, and disposal facilities. DOE began offsite shipments of hazardous wastes to treatment and disposal facilities in 1987. In 1990, DOE imposed a moratorium on shipments of hazardous waste that came from radiologically controlled areas or that had not been TE

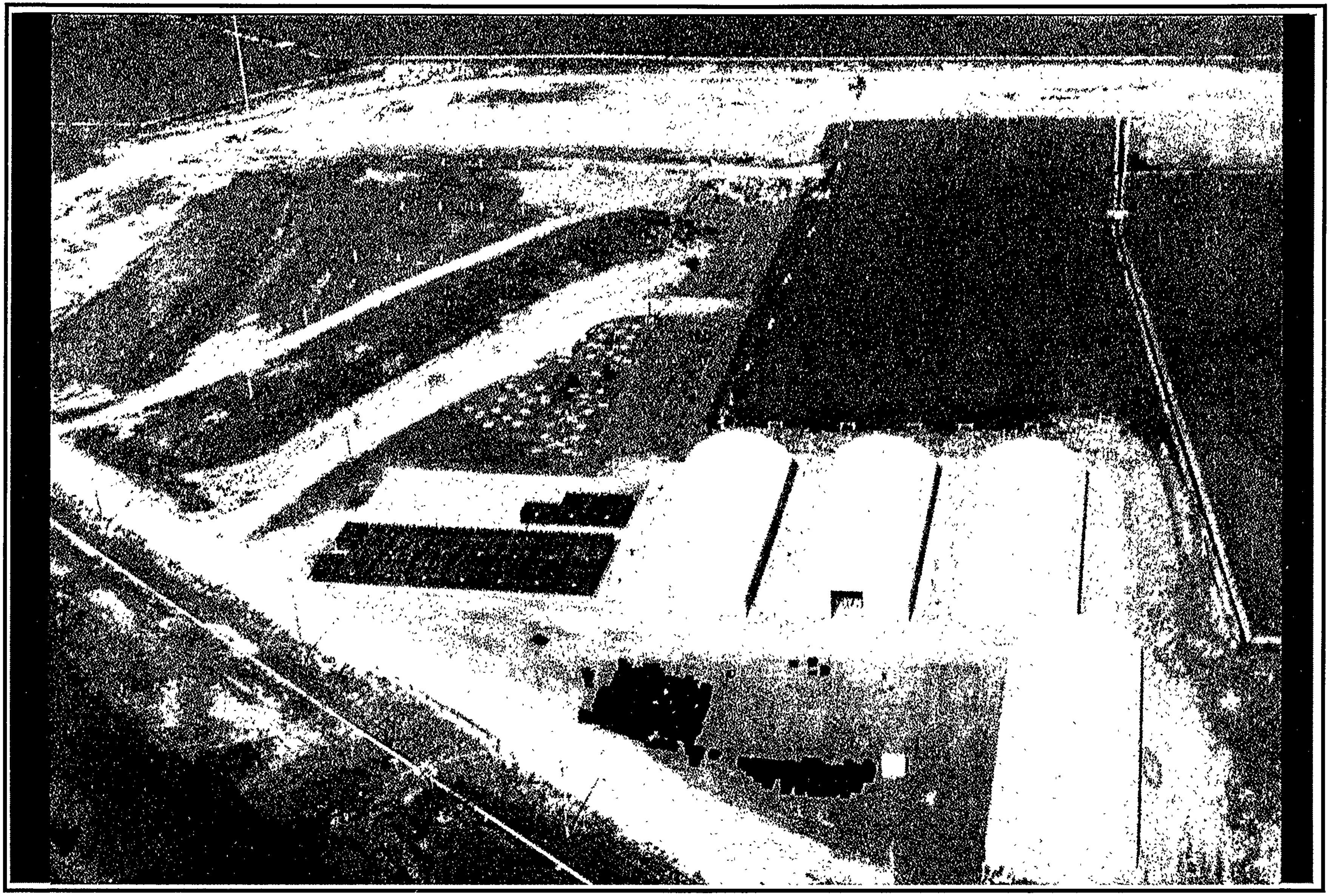

Figure C-32. Transuranic waste storage (E-Area).

TE

proven to be nonradioactive. The SRS continues to ship hazardous waste that is validated as nonradioactive waste (e.g., recyclable solvents) off the Site for recycling, treatment, or disposal.

The SRS mixed waste program consists primarily of safe storage until treatment and disposal facilities are available.

C.15.5 LOW-LEVEL LIQUID WASTES

The F- and H-Area Effluent Treatment Facility (ETF) decontaminates and treats low-level process water and stormwater contaminated with radioactive and chemical constituents. Routine influents accepted by the ETF are primarily evaporator condensates from the chemical separations facilities and the tank farms. Approximately 34 percent of the influent to the F- and H-Area ETF comes from F-Area, including the separations facility, cooling and stormwater retention basins, evaporator overheads, and laboratory liquid waste. H-Area influents comprise approximately 48 percent of the influents and include the separations facility, cooling and stormwater retention basins, evaporator condensate, tritium laboratory liquid waste, water inside the In-Tank Precipitation dike (an embankment designed to control water runoff), and laboratory liquid waste. The remainder comes from other F- and H-Area facilities.

The F and H-Area Effluent Treatment Facility was built to replace the old F- and H-Area seepage basins, which, under the 1984 Hazardous and Solid Waste Amendments to the Resource Conservation and Recovery Act, could not be used after 1988. The F- and H-Area ETF began operations in October 1988.

The F- and H-Area Effluent Treatment Facility decontaminates wastewater through a series of steps consisting of pH adjustment, sub-micron filtration, heavy-metal and organic adsorption, reverse osmosis, and ion exchange. The treatment steps concentrate the contaminants into a smaller volume of secondary waste, which is concentrated further by evaporation. The waste concentrate is eventually disposed of in the Z-Area Saltstone Manufacturing and Disposal Facility. The treated effluent is analyzed to ensure that it has been properly decontaminated and discharged to Upper Three Runs Creek.

APPENDIX D

ANNUAL DATA FOR PHASES ASSOCIATED WITH THE MANAGEMENT OF MATERIALS

TABLE OF CONTENTS

LIST OF TABLES (continued)

Table Page

D-30 Data for impact analysis of processing plutonium and uranium stored in vaults to oxide D-37
D-31 Data for impact analysis of processing and storage for vitrification (DWPF) of plutonium and uranium stored in vaults D-38
D-32 Data for impact analysis of vitrification (F-Canyon) of plutonium and uranium stored in vaults D-39
D-33 Data for impact analysis of continuing storage (No Action) of plutonium and uranium stored in vaults D-40
D-34 Data for impact analysis of improving storage of plutonium and uranium stored in vaults (plutonium-238 scrap material) D-41
D-35 Data for impact analysis of processing plutonium and uranium stored in vaults (plutonium-238 scrap material) to oxide D-42
D-36 Data for impact analysis of processing and storage for vitrification (DWPF) of plutonium and uranium stored in vaults (plutonium-238 scrap material) D-43
D-37 Data for impact analysis of continuing storage (No Action) of plutonium and uranium stored in vaults (plutonium-238 scrap material) D-44
D-38 Data for impact analysis of processing Mark-31 targets to metal D-45
D-39 Data for impact analysis of processing Mark-31 targets to oxide D-46
D-40 Data for impact analysis of improving storage of Mark-31 targets D-47
D-41 Data for impact analysis of improving storage (accelerated schedule) of Mark-31 targets D-48
D-42 Data for impact analysis of processing and storage for vitrification (DWPF) of Mark-31 targets D-49
D-43 Data for impact analysis of vitrification (F-Canyon) of Mark-31 targets D-50
D-44 Data for impact analysis of continuing storage (No Action) of Mark-31 targets D-51
D-45 Data for impact analysis of blending Mark 16/22 fuels to low enriched uranium D-52
D-46 Data for impact analysis of processing Mark 16/22 fuels to oxide D-53
D-47 Data for impact analysis of improving storage of Mark 16/22 fuels D-54
D-48 Data for impact analysis of improving storage (accelerated schedule) of Mark 16/22 fuels D-55
D-49 Data for impact analysis of processing and storage for vitrification (DWPF) of Mark 16/22 fuels D-56
D-50 Data for impact analysis of continuing storage (No Action) of Mark 16/22 fuels D-57
D-51 Data for impact analysis of processing and storage for vitrification (DWPF) of other aluminum-clad targets D-58
D-52 Data for impact analysis of improving storage of other aluminum-clad targets D-59
D-53 Data for impact analysis of improving storage (accelerated schedule) of other aluminum-clad targets D-60
D-54 Data for impact analysis of continuing storage (No Action) of other aluminum-clad targets D-61
D-55 Data for impact analysis of continuing storage (No Action) of Taiwan Research Reactor and Experimental Breeder Reactor-II fuel D-62
D-56 Data for impact analysis of processing Taiwan Research Reactor and Experimental Breeder Reactor-II fuel to metal D-63
D-57 Data for impact analysis of processing Taiwan Research Reactor and Experimental Breeder Reactor-II fuel to oxide D-64

TABLE OF CONTENTS

LIST OF TABLES (continued)

Table | Page

D-58 Data for impact analysis of processing and storage for vitrification (DWPF) of Taiwan Research Reactor and Experimental Breeder Reactor-II fuel D-65
D-59 Data for impact analysis of vitrification (F-Canyon) of Taiwan Research Reactor and Experimental Breeder Reactor-II fuel D-66
D-60 Data for impact analysis of improving storage of Taiwan Research Reactor and Experimental Breeder Reactor-II fuel D-67
D-61 Data for impact analysis of improving storage (accelerated schedule) of Taiwan Research Reactor and Experimental Breeder Reactor-II fuel D-68

APPENDIX D. ANNUAL DATA FOR PHASES ASSOCIATED WITH THE MANAGEMENT OF MATERIALS

This appendix contains the annual data used to calculate 10-year impacts from the various alternatives (see Chapter 4). Most of the alternatives would involve the use of multiple facilities and sequential steps to achieve the primary objective (i.e., stabilization or a form that satisfied program requirements). DOE estimated the annual impacts that could occur for each step or "phase" of each alternative. DOE then estimated durations for each phase of each alternative to generate the 10-year data.

This EIS uses the following generic names for the phases to facilitate the presentation of data, even though the different alternatives would involve different activities and facilities.

- **Existing Storage:** Actions associated with storing the material in its present form and configuration.

- **Characterization:** Actions that would be necessary to prepare the material for conversion, including visual inspection, weighing, and chemical and radioactive analysis. The characterization of the material would be needed to determine the implementation of proper processing technique(s).

- **Conversion:** Actions associated with changing the physical or chemical form of the material (i.e., liquid to solid). This typically would involve transfer of the material to a chemical processing facility and operation of the facility.

- **Interim Storage:** For some alternatives, the initial conversion or processing would not complete the stabilization process. Additional steps could be required, such as special packaging or further separations operations. Interim storage would include actions associated with storing the material in preparation for the next phase.

- **Additional Conversion:** Any additional actions necessary to place the material in a suitable form for continued storage, such as heating or repackaging solid forms of plutonium.

- **Packaging/Repackaging:** Actions necessary to place the suitable material form into an acceptable storage configuration, such as treatment in the Actinide Packaging Facility or repackaging.

- **Post-Stabilization Storage:** Actions associated with the material after it had been placed in a configuration and facility suitable for an extended storage period.

Table D-1 presents general information on actions associated with the phases for the alternatives that DOE considered for each material. In this table, "NA" indicates that a phase does not apply to an alternative.

The description of alternatives in Chapter 2 presents the projected durations for "active" phases (i.e., phases that would not involve storage) for the stabilization alternatives. For some alternatives, the latter phases are not likely to be completed by the end of the 10-year period analyzed in this EIS. For a few alternatives, the latter phases would not start within the 10-year period. Chapter 4 presents the impacts estimated for the next 10 years for various combinations of alternatives. To ensure a complete analysis, Tables D-2 through D-61 present the estimated annual impacts for each phase of every alternative, even though some are not likely to occur within the 10-year period. To the extent practical, DOE used historic data to predict and estimate future impacts. The data in this appendix are adapted (i.e., multiplied by conversion factors so the parameters could be expressed in metric units) from data provided by Westinghouse Savannah River Company (WSRC 1995a), except as noted. In general, the highest annual data are related to the conversion phase, because this phase would involve the transfer of the nuclear material and the operation of major facilities. The values for post-stabilization storage are generally less than those for existing storage, reflecting changes in material properties or storage configuration. TC

Table D-1. Actions associated with each phase.

Material	Alternative	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage	
Stable material	Continuing Storage (No Action)	Store in various SRS locations	NA[a]	NA	NA	NA	NA	NA	
Plutonium-242	Processing to Oxide	H-Canyon storage	NA	H-Canyon and HB-Line operation	Vault in F-Area	NA	Actinide Packaging Facility	F-Area	
	Vitrification (F-Canyon)	H-Canyon storage	NA	H-Canyon and HB-Line operation	Vault in F-Area	Vitrification (F-Canyon)	NA	F-Area vault	
	Processing and Storage for Vitrification (DWPF)[b]	H-Canyon storage	NA	H-Canyon operation	Tank farm storage	Vitrification (DWPF)	NA	Storage of glass cylinders	TC
	Processing to metal	H-Canyon storage	NA	H-Canyon transfer operations and F-Canyon receipt operation	NA	FB-Line operation	Actinide Packaging Facility	New vault storage	
	Continuing Storage (No Action)	H-Canyon storage	NA	NA	NA	NA	NA	NA	
Americium and curium	Vitrification (F-Canyon)	F-Canyon storage	NA	F-Canyon operation and vitrification	F-Area storage	NA	NA	NA	
	Processing to Oxide	F-Canyon storage	NA	F-Canyon operation and conversion to oxide	F-Area storage	NA	NA	NA	
	Processing and Storage for Vitrification (DWPF)	F-Canyon storage	NA	F-Canyon operation	Tank farm storage	Vitrification (DWPF)	NA	Storage of glass cylinders	TC
	Continuing Storage (No Action)	F-Canyon Storage	NA	NA	NA	NA	NA	NA	
Americium and curium targets	No Action	Storage in basins	NA	NA	NA	NA	NA	NA	
	Vitrification (F-Canyon)	Storage in basins	NA	Fuel transfer and F-Canyon operation	F-Canyon vault	NA	NA	NA	TC
	Processing to Oxide	Storage in basins	NA	Fuel transfer and F-Canyon operation	F-Canyon vault	NA	NA	NA	

Table D-1. (continued).

Material	Alternative	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage	
Neptunium-237	Processing to Oxide	H-Canyon storage	NA	H-Canyon and HB-Line operation	NA	NA	Actinide Packaging Facility	F-Area	
	Vitrification (F-Canyon)	H-Canyon storage	NA	H-Canyon transfer operations and F-Canyon receipt operations	NA	F-Canyon operation and vitrification	NA	F-Area	
	Processing and Storage for Vitrification (DWPF)	H-Canyon storage	NA	H-Canyon operation	Tank farm storage	Vitrification (DWPF)	NA	Storage of glass cylinders	TC
	Continuing Storage (No Action)	H-Canyon storage	NA	NA	NA	NA	NA	NA	
H-Canyon plutonium-239 solutions	Processing to Oxide	H-Canyon storage	NA	H-Canyon and HB-Line operation	Vault in F-Area	NA	Actinide Packaging Facility	F-Area	
	Processing and Storage for Vitrification (DWPF)	H-Canyon storage	NA	H-Canyon operation and transfer to tank farm	Tank farm storage	Vitrification (DWPF)	NA	Storage of glass cylinders	TE
	Vitrification (F-Canyon)	H-Canyon storage	NA	H-Canyon transfer operations and F-Canyon receipt operations	NA	F-Canyon operation and vitrification	NA	F-Area	
	Processing to Metal	H-Canyon storage	NA	H-Canyon transfer operation and F-Canyon receipt operation	NA	FB-Line operations	Actinide Packaging Facility	F-Area	
	Continuing Storage	H-Canyon storage	NA	NA	NA	NA	NA	NA	

Table D-1. (continued).

Material	Alternative	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
H-Canyon enriched uranium solutions	Blending Down to Low Enriched Uranium	H-Canyon storage	NA	H-Canyon operation	HA-Line	FA-Line operation	NA	F-Area
	Processing to Oxide (USF)	H-Canyon storage	NA	H-Canyon	HA-Line	USF operation	NA	H-Area
	Processing and Storage for Vitrification (DWPF)	H-Canyon storage	NA	H-Canyon operation and transfer to tank farm	Tank farm storage	Vitrification (DWPF)	NA	Storage of glass cylinders
	Continuing Storage (No Action)	H-Canyon storage	NA	NA	NA	NA	NA	NA
Plutonium and uranium stored in vaults	Improving Storage	F-Area vault storage	FB-Line or HB-Line characterization	NA	Vault in F-Area	NA	Actinide Packaging Facility	F-Area
	Processing to Oxide	F-Area vault storage	FB-Line or HB-Line characterization	H-Canyon and HB-Line or F-Canyon and FB-Line	Vault in F-Area	NA	Actinide Packaging Facility	F-Area
	Processing to Metal	F-Area vault storage	FB-Line or HB-Line characterization	F-Canyon and FB-Line operation	Vault in F-Area	NA	Actinide Packaging Facility	F-Area
	Processing and Storage for Vitrification (DWPF)	F-Area vault storage	NA	Transfer to tank farm via H-Canyon/HB-Line or F-Canyon/ FB-Line	Tank farm storage	Vitrification (DWPF)	NA	Storage of glass cylinders
	Vitrification (F-Canyon)	F-Area vault storage	FB-Line or HB-Line characterization	F-Canyon/FB-Line operation	NA	Vitrification (F-Canyon)	NA	F-Area
	Continuing Storage (No Action)	F-Area vault storage	NA	NA	NA	NA	NA	NA
Plutonium-238	Improving Storage	H-Area vault storage	NA	NA	H-Area	NA	NA	NA

TE

TE

Table D-1. (continued).

Material	Alternative	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
	Processing to Oxide	H-Area vault storage	NA	H-Canyon and HB-Line operation	H-Area	NA	NA	NA
	Processing and Storage for Vitrification (DWPF)	H-Area vault storage	NA	Transfer to tank farm via H-Canyon/HB-Line	Tank farm storage	Vitrification (DWPF)	NA	Storage of glass cylinders
	Continuing Storage (No Action)	H-Area vault storage	NA	NA	NA	NA	NA	NA
Mark-31 targets	Processing to Metal	Reactor basin storage	NA	Reactor fuel transfer and F-Canyon/ FB-Line operation	Vault in F-Area	NA	Actinide Packaging Facility	F-Area
	Processing to Oxide	Reactor basin storage	NA	Reactor fuel transfer and F-Canyon/FB-Line operation	Vault in F-Area	NA	Actinide Packaging Facility	F-Area
	Improving Storage	Reactor basin storage	NA	NA	Reactor basin storage	NA	Package for dry storage	Dry Storage Facility
	Processing and Storage for Vitrification (DWPF)	Reactor basin storage	NA	Reactor fuel transfer and transfer to tank farm via F-Canyon	Tank farm storage	Vitrification (DWPF)	NA	Storage of glass cylinders
	Vitrification (F-Canyon)	Reactor basin storage	NA	Reactor fuel transfer and F-Canyon/FB-Line operation	Vault in F-Area	Vitrification (F-Canyon)	NA	F-Area
	Continuing Storage (No Action)	Reactor basin storage	NA	NA	Reactor basin storage	NA	NA	NA

Table D-1. (continued).

Material	Alternative	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Mark-16 and -22 fuels	Blending Down to Low Enriched Uranium	Reactor basin storage	NA	Reactor fuel transfer and F-Canyon/ FA-Line operation or H-Canyon/ HA-Line operation	HA-Line (if initially converted in H-Area) or FA-Line (if converted in F-Area)	FA-Line operation	NA	F-Area
	Processing to Oxide (USF)	Reactor basin storage	NA	Reactor fuel transfer and H-Canyon operation	HA-Line	USF operation	NA	H-Area
	Improving Storage	Reactor basin storage	NA	NA	Reactor basin storage	NA	Package for dry storage	Dry Storage Facility
	Processing and Storage for Vitrification (DWPF)	Reactor basin storage	NA	Reactor fuel transfer and transfer to tank farm	Tank farm storage	Vitrification (DWPF)	NA	Storage of glass cylinders
	Continuing Storage (No Action)	Reactor basin storage	NA	NA	NA	NA	NA	NA
Other aluminum-clad targets	Processing and Storage for Vitrification (DWPF)	Reactor basin storage	NA	Reactor fuel transfer and transfer to tank farm	Tank farm storage	Vitrification (DWPF)	NA	Storage of glass cylinders
	Improving Storage	Reactor basin storage	NA	NA	Reactor basin storage	NA	Package for dry storage	Dry Storage Facility
	Continuing Storage (No Action)	Reactor basin storage	NA	NA	NA	NA	NA	NA

Table D-1. (continued).

Material	Alternative	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
TRR fuel and EBR-II slugs[c]	Continuing Storage (No Action)	Receiving Basin for Offsite Fuel	NA	NA	NA	NA	NA	NA
	Processing to Metal	Receiving Basin for Offsite Fuel	NA	Fuel Transfer and F-Canyon/FB-Line operation	Vault in F-Area	NA	Actinide Packaging Facility	F-Area
	Processing to Oxide	Receiving Basin for Offsite Fuel	NA	Fuel Transfer and F-Canyon/FB-Line operation	Vault in F-Area	NA	Actinide Packaging Facility	F-Area
	Processing and Storage for Vitrification (DWPF)	Receiving Basin for Offsite Fuel	NA	Fuel Transfer and transfer to tank farm	Tank farm storage	Vitrification (DWPF)	NA	Storage of glass cylinders
	Vitrification (F-Canyon)	Receiving Basin for Offsite Fuel	NA	Fuel Transfer and F-Canyon/FB-Line operation	Vault in F-Area	Vitrification (F-Canyon)	NA	F-Area
	Improving Storage	Receiving Basin for Offsite Fuel	NA	NA	Receiving Basin for Offsite Fuel	NA	Package for dry storage	Dry Storage Facility

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a. NA = Not applicable.

b. DWPF = Defense Waste Processing Facility.

c. TRR = Taiwan Research Reactor; EBR = Experimental Breeder Reactor.

TC

Table D-2. Data for impact analysis of continuing storage (No Action) of stable materials.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	1.9×10^{-6}	NA	NA	NA	NA	NA	NA
Liquid MEI dose (rem/year)	8.4×10^{-7}	NA	NA	NA	NA	NA	NA
Total MEI dose (rem/year)	2.7×10^{-6}	NA	NA	NA	NA	NA	NA
Atmospheric population dose (person-rem/year)[c]	0.10	NA	NA	NA	NA	NA	NA
Liquid population dose (person-rem/year)[c]	1.6×10^{-2}	NA	NA	NA	NA	NA	NA
Total population dose (person-rem/year)	0.12	NA	NA	NA	NA	NA	NA
Offsite CO concentration (μg/m^3) - 1-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite CO concentration (μg/m^3) - 8-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite NO_x concentration (μg/m^3) - annual average[d]	5.32×10^{-2}	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration (μg/m^3) - 3-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration (μg/m^3) - 24-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration (μg/m^3) - annual average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 12-hour average[d]	9.86×10^{-3}	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 24-hour average[d]	5.31×10^{-3}	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 1-week average[d]	2.09×10^{-3}	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 1-month average[d]	5.69×10^{-4}	NA	NA	NA	NA	NA	NA
Offsite HNO_3 concentration (μg/m^3) - 24-hour average[d]	0.147	NA	NA	NA	NA	NA	NA
Offsite HNO_3 concentration (μg/m^3) - annual average[d]	9.93×10^{-3}	NA	NA	NA	NA	NA	NA
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	NA	NA	NA	NA
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	3.36×10^{-2}	NA	NA	NA	NA	NA	NA
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	NA	NA	NA	NA
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	2.3×10^{-3}	NA	NA	NA	NA	NA	NA
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	NA	NA	NA	NA
Average number of radiation workers	410	NA	NA	NA	NA	NA	NA
Collective worker dose (person-rem/year)	14	NA	NA	NA	NA	NA	NA
Water usage (millions of liters/year)	1,629	NA	NA	NA	NA	NA	NA
Electricity usage (megawatt-hour/year)	39,528	NA	NA	NA	NA	NA	NA
Steam usage (millions of kilograms/year)	159	NA	NA	NA	NA	NA	NA
Fuel usage (thousands of liters/year)	960	NA	NA	NA	NA	NA	NA
High-level liquid waste generation (millions of liters per year)	2.1	NA	NA	NA	NA	NA	NA
Equivalent DWPF canisters (per year)	4	NA	NA	NA	NA	NA	NA
Saltstone generation (cubic meters/year)	1,083	NA	NA	NA	NA	NA	NA
TRU waste generation (cubic meters/year)	2	NA	NA	NA	NA	NA	NA
Hazardous/mixed waste generation (cubic meters/year)	6	NA	NA	NA	NA	NA	NA
Low-level waste generation (cubic meters/year)	4,097	NA	NA	NA	NA	NA	NA

TC

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

Table D-3. Data for impact analysis of processing plutonium-242 solutions to oxide.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	9.4×10^{-7}	NA	1.7×10^{-4}	1.1×10^{-9}	NA	6.6×10^{-9}	4.4×10^{-9}
Liquid MEI dose (rem/year)	3.7×10^{-6}	NA	6.2×10^{-6}	1.1×10^{-10}	NA	7.3×10^{-10}	4.3×10^{-10}
Total MEI dose (rem/year)	4.7×10^{-6}	NA	1.7×10^{-4}	1.2×10^{-9}	NA	7.3×10^{-9}	4.8×10^{-9}
Atmospheric population dose (person-rem/year)[c]	3.8×10^{-2}	NA	6.8	5.0×10^{-5}	NA	2.9×10^{-4}	2.0×10^{-4}
Liquid population dose (person-rem/year)[c]	1.1×10^{-2}	NA	1.9×10^{-2}	1.7×10^{-6}	NA	6.4×10^{-6}	6.8×10^{-6}
Total population dose (person-rem/year)	4.9×10^{-2}	NA	6.8	5.2×10^{-5}	NA	3.0×10^{-4}	2.0×10^{-4}
Offsite CO concentration (μg/m^3) - 1-hour average[d]	1.30	NA	3.96	0	NA	0	0
Offsite CO concentration (μg/m^3) - 8-hour average[d]	0.311	NA	9.47	0	NA	0	0
Offsite NO_x concentration (μg/m^3) - annual average[d]	1.2×10^{-2}	NA	3.26×10^{-2}	0	NA	0	0
Offsite SO_2 concentration (μg/m^3) - 3-hour average[d]	7.54×10^{-4}	NA	2.32×10^{-3}	0	NA	0	0
Offsite SO_2 concentration (μg/m^3) - 24-hour average[d]	1.70×10^{-4}	NA	5.21×10^{-4}	0	NA	0	0
Offsite SO_2 concentration (μg/m^3) - annual average[d]	1.07×10^{-5}	NA	3.27×10^{-5}	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 12-hour average[d]	0	NA	0	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 24-hour average[d]	0	NA	0	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 1-week average[d]	0	NA	0	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 1-month average[d]	0	NA	0	0	NA	0	0
Offsite HNO_3 concentration (μg/m^3) - 24-hour average[d]	3.21×10^{-3}	NA	8.74×10^{-3}	0	NA	0	0
Offsite HNO_3 concentration (μg/m^3) - annual average[d]	2.01×10^{-4}	NA	5.49×10^{-4}	0	NA	0	0
Onsite CO concentration (mg/m^3) - 8-hour average[d]	2.08×10^{-3}	NA	6.32×10^{-3}	0	NA	0	0
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	6.32×10^{-3}	NA	1.72×10^{-2}	0	NA	0	0
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	2.99×10^{-6}	NA	9.19×10^{-6}	0	NA	0	0
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	5.65×10^{-5}	NA	1.54×10^{-4}	0	NA	0	0
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	1.47×10^{-6}	NA	4.57×10^{-6}	0	NA	0	0
Average number of radiation workers	134	NA	291	24	NA	157	80
Collective worker dose (person-rem/year)	1.34	NA	36	9.8×10^{-1}	NA	31	16
Water usage (millions of liters/year)	174	NA	212	139	NA	500	454
Electricity usage (megawatt-hour/year)	13,299	NA	19,591	1,842	NA	6,620	6,018
Steam usage (millions of kilograms/year)	79	NA	84	2	NA	7	6
Fuel usage (thousands of liters/year)	499	NA	520	6	NA	21	19
High-level liquid waste generation (millions of liters per year)	0.12	NA	0.12	0	NA	0	0
Equivalent DWPF canisters (per year)	2	NA	2	0	NA	0	0
Saltstone generation (cubic meters/year)	325	NA	325	0	NA	0	0
TRU waste generation (cubic meters/year)	0	NA	98	1	NA	1	0
Hazardous/mixed waste generation (cubic meters/year)	0	NA	119	0	NA	1	0
Low-level waste generation (cubic meters/year)	556	NA	1,326	484	NA	7	5

TC

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

Table D-4. Data for impact analysis of vitrification (F-Canyon) of plutonium-242 solutions.[a,b]

	Existing storage	Analyze/ pretreat	Process	Interim storage	Final stabilization	Storage preparation	Long-term storage
Atmospheric MEI dose (rem/yr)	9.4×10^{-7}	NA	1.7×10^{-4}	1.1×10^{-9}	4.3×10^{-6}	NA	4.4×10^{-9}
Liquid MEI dose (rem/yr)	3.7×10^{-6}	NA	6.2×10^{-6}	1.1×10^{-10}	1.3×10^{-7}	NA	4.3×10^{-10}
Total MEI dose (rem/yr)	4.7×10^{-6}	NA	1.7×10^{-4}	1.2×10^{-9}	4.4×10^{-6}	NA	4.8×10^{-9}
Atmospheric population dose (person-rem/yr)[c]	3.8×10^{-2}	NA	6.8	5.0×10^{-5}	0.19	NA	2.0×10^{-4}
Liquid population dose (person-rem/yr)[c]	1.1×10^{-2}	NA	1.9×10^{-2}	1.7×10^{-6}	7.3×10^{-4}	NA	6.8×10^{-6}
Total population dose (person-rem/yr)	4.9×10^{-2}	NA	6.8	5.2×10^{-5}	0.19	NA	2.0×10^{-4}
Offsite CO concentration (μg/m^3) - 1 hour average[d]	1.30	NA	3.96	0	0	NA	0
Offsite CO concentration (μg/m^3) - 8 hour average[d]	0.311	NA	0.947	0	0	NA	0
Offsite NO_x concentration (μg/m^3) - annual average[d]	1.20×10^{-2}	NA	3.26×10^{-2}	0	0.113	NA	0
Offsite SO_2 concentration (μg/m^3) - 3 hour average[d]	7.54×10^{-4}	NA	2.32×10^{-3}	0	0	NA	0
Offsite SO_2 concentration (μg/m^3) - 24 hour average[d]	1.70×10^{-4}	NA	5.21×10^{-4}	0	0	NA	0
Offsite SO_2 concentration (μg/m^3) - annual average[d]	1.07×10^{-5}	NA	3.27×10^{-5}	0	0	NA	0
Offsite Gaseous Fluorides (μg/m^3) 12-hour average[d]	0	NA	0	0	2.1×10^{-2}	NA	0
Offsite Gaseous Fluorides (μg/m^3) 24-hour average[d]	0	NA	0	0	1.13×10^{-2}	NA	0
Offsite Gaseous Fluorides (μg/m^3) 1-week average[d]	0	NA	0	0	4.43×10^{-3}	NA	0
Offsite Gaseous Fluorides (μg/m^3) 1-month average[d]	0	NA	0	0	1.25×10^{-3}	NA	0
Offsite HNO_3 concentration (μg/m^3) - 24-hour average[d]	3.21×10^{-3}	NA	8.74×10^{-3}	0	3.11×10^{-1}	NA	0
Offsite HNO_3 concentration (μg/m^3) - annual average[d]	2.01×10^{-4}	NA	5.49×10^{-4}	0	2.11×10^{-2}	NA	0
Onsite CO concentration (mg/m^3) - 8 hour average[d]	2.08×10^{-3}	NA	6.32×10^{-3}	0	0	NA	0
Onsite NO_x concentration (mg/m^3) - 1 hour average[d]	6.32×10^{-3}	NA	1.72×10^{-2}	0	7.15×10^{-2}	NA	0
Onsite SO_2 concentration (mg/m^3) - 8 hour average[d]	2.99×10^{-6}	NA	9.19×10^{-6}	0	0	NA	0
Onsite HNO_3 concentration (mg/m^3) - 8 hour average[d]	5.65×10^{-5}	NA	1.54×10^{-4}	0	4.89×10^{-3}	NA	0
Onsite CO_2 concentration (mg/m^3) - 8 hour average[d]	1.47×10^{-6}	NA	4.57×10^{-6}	0	0	NA	0
Average Number of radiation workers	134	NA	291	24	463	NA	80
Collective worker dose (person-rem/year)	1.34	NA	36	0.98	120	NA	16
Water usage (millions of liters/year)	174	NA	212	139	600	NA	454
Electricity usage (megawatt-hour/year)	13,299	NA	19,591	1,842	11,088	NA	6,018
Steam usage (millions of kilograms/year)	79	NA	84	2	54	NA	6
Fuel usage (thousands of liters/year)	499	NA	520	6	354	NA	19
High-level liquid waste generation (millions of liters per year)	0.12	NA	0.12	0	0.44	NA	0
Equivalent DWPF[f] canisters (per year)	2	NA	2	0	9	NA	0
Saltstone generation (cubic meters/year)	325	NA	325	0	1,189	NA	0
TRU[g] Waste generation (cubic meters/year)	0	NA	98	1	64	NA	0
Hazardous/Mixed waste generation (cubic meters/year)	0	NA	119	0	0	NA	0
Low-Level Waste generation (cubic meters/year)	556	NA	1,326	484	4,986	NA	5

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-5. Data for impact analysis of processing plutonium-242 solutions to metal.[a,b]

Factor	Existing storage	Analyze/ pretreat	Process	Interim storage[a]	Final stabilization	Storage preparation	Long-term storage[b]
Atmospheric MEI dose (rem/year)	9.4×10^{-7}	NA	2.8×10^{-6}	NA	4.0×10^{-6}	6.6×10^{-9}	4.4×10^{-9}
Liquid MEI dose (rem/year)	3.7×10^{-6}	NA	3.9×10^{-6}	NA	2.4×10^{-7}	7.3×10^{-10}	4.3×10^{-10}
Total MEI dose (rem/year)	4.7×10^{-6}	NA	6.7×10^{-6}	NA	4.2×10^{-6}	7.3×10^{-9}	4.8×10^{-9}
Atmospheric population dose (person-rem/year)[c]	3.8×10^{-2}	NA	0.12	NA	0.18	2.9×10^{-4}	2.0×10^{-4}
Liquid population dose (person-rem/year)[c]	1.1×10^{-2}	NA	1.4×10^{-2}	NA	3.2×10^{-3}	6.4×10^{-6}	6.8×10^{-6}
Total population dose (person-rem/year)	4.9×10^{-2}	NA	0.14	NA	0.18	3.0×10^{-4}	2.0×10^{-4}
Offsite CO concentration (μg/m^3) - 1-hour averaged[d]	1.3	NA	2.16	NA	0	0	0
Offsite CO concentration (μg/m^3) - 8-hour averaged[d]	0.311	NA	0.444	NA	0	0	0
Offsite NO_x concentration (μg/m^3) - annual averaged[d]	1.2×10^{-2}	NA	0.101	NA	0.140	0	0
Offsite SO_2 concentration (μg/m^3) - 3-hour averaged[d]	7.54×10^{-4}	NA	1.31×10^{-3}	NA	0	0	0
Offsite SO_2 concentration (μg/m^3) - 24-hour averaged[d]	1.7×10^{-4}	NA	2.30×10^{-4}	NA	0	0	0
Offsite SO_2 concentration (μg/m^3) - annual averaged[d]	1.07×10^{-5}	NA	1.56×10^{-5}	NA	0	0	0
Offsite gaseous fluorides (μg/m^3) 12-hour averaged[d]	0	NA	1.55×10^{-2}	NA	2.60×10^{-2}	0	0
Offsite gaseous fluorides (μg/m^3) 24-hour averaged[d]	0	NA	8.35×10^{-3}	NA	1.4×10^{-2}	0	0
Offsite gaseous fluorides (μg/m^3) 1-week averaged[d]	0	NA	3.28×10^{-3}	NA	5.5×10^{-3}	0	0
Offsite gaseous fluorides (μg/m^3) 1-month averaged[d]	0	NA	9.25×10^{-4}	NA	1.55×10^{-3}	0	0
Offsite HNO_3 concentration (μg/m^3) - 24-hour averaged[d]	3.21×10^{-3}	NA	0.234	NA	0.385	0	0
Offsite HNO_3 concentration (μg/m^3) - annual averaged[d]	2.01×10^{-4}	NA	1.59×10^{-2}	NA	2.62×10^{-2}	0	0
Onsite CO concentration (mg/m^3) - 8-hour averaged[d]	2.08×10^{-3}	NA	2.5×10^{-3}	NA	0	0	0
Onsite NO_x concentration (mg/m^3) - 1-hour averaged[d]	6.32×10^{-3}	NA	6.4×10^{-2}	NA	8.87×10^{-2}	0	0
Onsite SO_2 concentration (mg/m^3) - 8-hour averaged[d]	2.99×10^{-6}	NA	3.61×10^{-6}	NA	0	0	0
Onsite HNO_3 concentration (mg/m^3) - 8-hour averaged[d]	5.65×10^{-5}	NA	3.68×10^{-3}	NA	6.06×10^{-3}	0	0
Onsite CO_2 concentration (mg/m^3) - 8-hour averaged[d]	1.47×10^{-6}	NA	1.77×10^{-6}	NA	0	0	0
Average number of radiation workers	134	NA	342	NA	589	157	80
Collective worker dose (person-rem/year)	1.3	NA	13	NA	120	31	16
Water usage (millions of liters/year)	174	NA	1,089	NA	1,216	500	454
Electricity usage (megawatt-hour/year)	13,299	NA	30,233	NA	22,478	6,620	6,018
Steam usage (millions of kilograms/year)	79	NA	162	NA	110	7	6
Fuel usage (thousands of liters/year)	499	NA	1,040	NA	718	21	19
High-level liquid waste generation (millions of liters/year)	0.12	NA	0.30	NA	0.62	0	0
Equivalent DWPF canisters (per year)	2	NA	6	NA	13	0	0
Saltstone generation (cubic meters/year)	325	NA	825	NA	1,689	0	0
TRU waste generation (cubic meters/year)	0	NA	0	NA	64	1	0
Hazardous/mixed waste generation (cubic meters/year)	0	NA	0	NA	0	1	0
Low-level waste generation (cubic meters/year)	556	NA	1,456	NA	5,771	7	5

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-6. Data for impact analysis of processing and storage for vitrification (DWPF) of plutonium-242 solutions.[a,b]

Factor	Existing storage	Analyze/pretreat	Process	Interim storage[c]	Final stabilization	Storage preparation	Long-term storage[d]
Atmospheric MEI dose (rem/year)	9.4×10^{-7}	NA	1.8×10^{-6}	-	1.6×10^{-4}	NA	-
Liquid MEI dose (rem/year)	3.7×10^{-6}	NA	7.1×10^{-6}	-	2.4×10^{-5}	NA	-
Total MEI dose (rem/year)	4.7×10^{-6}	NA	8.9×10^{-6}	-	1.8×10^{-4}	NA	-
Atmospheric population dose (person-rem/year)[e]	3.8×10^{-2}	NA	7.3×10^{-2}	-	6.9	NA	-
Liquid population dose (person-rem/year)[e]	1.1×10^{-2}	NA	2.1×10^{-2}	-	1.4×10^{-1}	NA	-
Total population dose (person-rem/year)	4.9×10^{-2}	NA	9.4×10^{-2}	-	7.0	NA	-
Offsite CO concentration (µg/m^3) - 1-hour average[f]	1.30	NA	5.85	-	0	NA	-
Offsite CO concentration (µg/m^3) - 8-hour average[f]	3.11×10^{-1}	NA	1.40	-	0	NA	-
Offsite NO_x concentration (µg/m^3) - annual average[f]	1.20×10^{-2}	NA	3.32×10^{-2}	-	0	NA	-
Offsite SO_2 concentration (µg/m^3) - 3-hour average[f]	7.54×10^{-4}	NA	3.41×10^{-3}	-	0	NA	-
Offsite SO_2 concentration (µg/m^3) - 24-hour average[f]	1.70×10^{-4}	NA	7.68×10^{-4}	-	0	NA	-
Offsite SO_2 concentration (µg/m^3) - annual average[f]	1.07×10^{-5}	NA	4.82×10^{-5}	-	0	NA	-
Offsite gaseous fluorides (µg/m^3) 12-hour average[f]	0	NA	7.48×10^{-6}	-	0	NA	-
Offsite gaseous fluorides (µg/m^3) 24-hour average[f]	0	NA	2.99×10^{-6}	-	0	NA	-
Offsite gaseous fluorides (µg/m^3) 1-week average[f]	0	NA	1.58×10^{-6}	-	0	NA	-
Offsite gaseous fluorides (µg/m^3) 1-month average[f]	0	NA	4.46×10^{-7}	-	0	NA	-
Offsite HNO_3 concentration (µg/m^3) - 24-hour average[f]	3.21×10^{-3}	NA	2.21×10^{-2}	-	0	NA	-
Offsite HNO_3 concentration (µg/m^3) - annual average[f]	2.01×10^{-4}	NA	1.39×10^{-3}	-	0	NA	-
Onsite CO concentration (mg/m^3) - 8-hour average[f]	2.08×10^{-3}	NA	9.33×10^{-3}	-	0	NA	-
Onsite NO_x concentration (mg/m^3) - 1-hour average[f]	6.32×10^{-2}	NA	1.76×10^{-2}	-	0	NA	-
Onsite SO_2 concentration (mg/m^3) - 8-hour average[f]	2.99×10^{-6}	NA	1.35×10^{-5}	-	0	NA	-
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[f]	5.65×10^{-5}	NA	3.90×10^{-4}	-	0	NA	-
Onsite CO_2 concentration (mg/m^3) - 8-hour average[f]	1.47×10^{-6}	NA	6.72×10^{-6}	-	0	NA	-
Average number of radiation workers	134	NA	304	-	1,500	NA	-
Collective worker dose (person-rem/year)	1.34	NA	3	-	60	NA	-
Water usage (millions of liters/year)	174	NA	280	-	1,476	NA	-
Electricity usage (megawatt-hour/year)	13,299	NA	21,450	-	50,371	NA	-
Steam usage (millions of kilograms/year)	79	NA	128	-	167	NA	-
Fuel usage (thousands of liters/year)	499	NA	805	-	0	NA	-
High-level liquid waste generation (millions of liters per year)	0.12	NA	0.23	-	0	NA	-
Equivalent DWPF canisters (per year)	2	NA	3	-	405	NA	-
Saltstone generation (cubic meters/year)	325	NA	619	-	0	NA	-
TRU waste generation (cubic meters/year)	0	NA	0	-	1,136	NA	-
Hazardous/mixed waste generation (cubic meters/year)	0	NA	0	-	3,055	NA	-
Low-level waste generation (cubic meters/year)	556	NA	686	-	2,731	NA	-

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Atmospheric releases from the waste tank farms in 1993 contributed less than 1×10^{-8} rem to the maximally exposed individual and less than 50 person-rem to the worker collective dose. At this time, DOE does not expect any discernible increase to these or other environmental factors as a result of this phase in the stabilization alternative.

d. DOE does not expect any discernible incremental impacts as a result of this phase in the stabilization alternative.

e. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from (1994a).

f. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-7. Data for impact analysis of continuing storage (No Action) of plutonium-242 solutions.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	9.4×10^{-7}	NA	NA	NA	NA	NA	NA
Liquid MEI dose (rem/year)	3.7×10^{-6}	NA	NA	NA	NA	NA	NA
Total MEI dose (rem/year)	4.7×10^{-6}	NA	NA	NA	NA	NA	NA
Atmospheric population dose (person-rem/year)[c]	3.8×10^{-2}	NA	NA	NA	NA	NA	NA
Liquid population dose (person-rem/year)[c]	1.1×10^{-2}	NA	NA	NA	NA	NA	NA
Total population dose (person-rem/year)	4.9×10^{-2}	NA	NA	NA	NA	NA	NA
Offsite CO concentration (μg/m^3) - 1-hour average[d]	1.30	NA	NA	NA	NA	NA	NA
Offsite CO concentration (μg/m^3) - 8-hour average[d]	0.311	NA	NA	NA	NA	NA	NA
Offsite NO_x concentration (μg/m^3) - annual average[d]	1.2×10^{-2}	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration (μg/m^3) - 3-hour average[d]	7.54×10^{-4}	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration (μg/m^3) - 24-hour average[d]	1.70×10^{-4}	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration (μg/m^3) - annual average[d]	1.07×10^{-5}	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 12-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 24-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 1-week average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 1-month average[d]	0	NA	NA	NA	NA	NA	NA
Offsite HNO_3 concentration (μg/m^3) - 24-hour average[d]	3.21×10^{-3}	NA	NA	NA	NA	NA	NA
Offsite HNO_3 concentration (μg/m^3) - annual average[d]	2.01×10^{-4}	NA	NA	NA	NA	NA	NA
Onsite CO concentration (mg/m^3) - 8-hour average[d]	2.08×10^{-3}	NA	NA	NA	NA	NA	NA
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	6.32×10^{-3}	NA	NA	NA	NA	NA	NA
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	2.99×10^{-6}	NA	NA	NA	NA	NA	NA
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	5.65×10^{-5}	NA	NA	NA	NA	NA	NA
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	1.47×10^{-6}	NA	NA	NA	NA	NA	NA
Average number of radiation workers	134	NA	NA	NA	NA	NA	NA
Collective worker dose (person-rem/year)	1.34	NA	NA	NA	NA	NA	NA
Water usage (millions of liters/year)	174	NA	NA	NA	NA	NA	NA
Electricity usage (megawatt-hour/year)	13,299	NA	NA	NA	NA	NA	NA
Steam usage (millions of kilograms/year)	79	NA	NA	NA	NA	NA	NA
Fuel usage (thousands of liters/year)	499	NA	NA	NA	NA	NA	NA
High-level liquid waste generation (millions of liters per year)	0.12	NA	NA	NA	NA	NA	NA
Equivalent DWPF canisters (per year)	2	NA	NA	NA	NA	NA	NA
Saltstone generation (cubic meters/year)	325	NA	NA	NA	NA	NA	NA
TRU waste generation (cubic meters/year)	0	NA	NA	NA	NA	NA	NA
Hazardous/mixed waste generation (cubic meters/year)	0	NA	NA	NA	NA	NA	NA
Low-level waste generation (cubic meters/year)	556	NA	NA	NA	NA	NA	NA

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-8. Data for impact analysis of vitrification (F-Canyon) of americium/curium solutions.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	1.5×10^{-6}	NA	1.1×10^{-5}	1.1×10^{-7}	NA	NA	NA
Liquid MEI dose (rem/year)	1.2×10^{-7}	NA	3.1×10^{-7}	3.9×10^{-9}	NA	NA	NA
Total MEI dose (rem/year)	1.6×10^{-6}	NA	1.1×10^{-5}	1.1×10^{-7}	NA	NA	NA
Atmospheric population dose (person-rem/year)[c]	6.7×10^{-2}	NA	0.48	4.8×10^{-3}	NA	NA	NA
Liquid population dose (person-rem/year)[c]	1.9×10^{-3}	NA	1.8×10^{-3}	6.1×10^{-5}	NA	NA	NA
Total population dose (person-rem/year)	6.9×10^{-2}	NA	0.48	4.8×10^{-3}	NA	NA	NA
Offsite CO concentration (μg/m^3) - 1-hour average[d]	0	NA	0	0	NA	NA	NA
Offsite CO concentration (μg/m^3) - 8-hour average[d]	0	NA	0	0	NA	NA	NA
Offsite NO_x concentration (μg/m^3) - annual average[d]	3.28×10^{-2}	NA	0.283	2.83×10^{-3}	NA	NA	NA
Offsite SO_2 concentration (μg/m^3) - 3-hour average[d]	0	NA	0	0	NA	NA	NA
Offsite SO_2 concentration (μg/m^3) - 24-hour average[d]	0	NA	0	0	NA	NA	NA
Offsite SO_2 concentration (μg/m^3) - annual average[d]	0	NA	0	0	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 12-hour average[d]	6.08×10^{-3}	NA	5.24×10^{-2}	5.24×10^{-4}	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 24-hour average[d]	3.27×10^{-3}	NA	2.82×10^{-2}	2.82×10^{-4}	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 1-week average[d]	1.29×10^{-3}	NA	1.11×10^{-2}	1.11×10^{-4}	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 1-month average[d]	3.63×10^{-4}	NA	3.12×10^{-3}	3.12×10^{-5}	NA	NA	NA
Offsite HNO_3 concentration (μg/m^3) - 24-hour average[d]	9.02×10^{-2}	NA	0.777	7.77×10^{-3}	NA	NA	NA
Offsite HNO_3 concentration (μg/m^3) - annual average[d]	6.12×10^{-3}	NA	5.27×10^{-2}	5.27×10^{-4}	NA	NA	NA
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	NA	NA	NA
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	2.07×10^{-2}	NA	0.179	1.79×10^{-3}	NA	NA	NA
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	NA	NA	NA
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	1.42×10^{-3}	NA	1.22×10^{-2}	1.22×10^{-4}	NA	NA	NA
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	NA	NA	NA
Average number of radiation workers	175	NA	409	4	NA	NA	NA
Collective worker dose (person-rem/year)	7	NA	64	0.64	NA	NA	NA
Water usage (millions of liters/year)	698	NA	1,499	15	NA	NA	NA
Electricity usage (megawatt-hour/year)	12,902	NA	27,722	277	NA	NA	NA
Steam usage (millions of kilograms/year)	63	NA	136	1	NA	NA	NA
Fuel usage (thousands of liters/year)	412	NA	886	9	NA	NA	NA
High-level liquid waste generation (millions of liters per year)	0.13	NA	1.4	0	NA	NA	NA
Equivalent DWPF canisters (per year)	3	NA	29	0	NA	NA	NA
Saltstone generation (cubic meters/year)	344	NA	3,888	0	NA	NA	NA
TRU waste generation (cubic meters/year)	0	NA	0	0	NA	NA	NA
Hazardous/mixed waste generation (cubic meters/year)	0	NA	0	0	NA	NA	NA
Low-level waste generation (cubic meters/year)	645	NA	1,276	6	NA	NA	NA

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-9. Data for impact analysis of processing americium/curium solutions to oxide.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	1.5×10^{-6}	NA	1.1×10^{-5}	1.1×10^{-7}	NA	NA	NA
Liquid MEI dose (rem/year)	1.2×10^{-7}	NA	3.1×10^{-7}	3.9×10^{-9}	NA	NA	NA
Total MEI dose (rem/year)	1.6×10^{-6}	NA	1.1×10^{-5}	1.1×10^{-7}	NA	NA	NA
Atmospheric population dose (person-rem/year)[c]	6.7×10^{-2}	NA	0.48	4.8×10^{-3}	NA	NA	NA
Liquid population dose (person-rem/year)[c]	1.9×10^{-3}	NA	1.8×10^{-3}	6.1×10^{-5}	NA	NA	NA
Total population dose (person-rem/year)	6.9×10^{-2}	NA	0.48	4.8×10^{-3}	NA	NA	NA
Offsite CO concentration ($\mu g/m^3$) - 1-hour averaged[d]	0	NA	0	0	NA	NA	NA
Offsite CO concentration ($\mu g/m^3$) - 8-hour averaged[d]	0	NA	0	0	NA	NA	NA
Offsite NO_x concentration ($\mu g/m^3$) - annual averaged[d]	3.28×10^{-2}	NA	0.283	2.83×10^{-3}	NA	NA	NA
Offsite SO_2 concentration ($\mu g/m^3$) - 3-hour averaged[d]	0	NA	0	0	NA	NA	NA
Offsite SO_2 concentration ($\mu g/m^3$) - 24-hour averaged[d]	0	NA	0	0	NA	NA	NA
Offsite SO_2 concentration ($\mu g/m^3$) - annual averaged[d]	0	NA	0	0	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 12-hour averaged[d]	6.08×10^{-3}	NA	5.24×10^{-2}	5.24×10^{-4}	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 24-hour averaged[d]	3.27×10^{-3}	NA	2.82×10^{-2}	2.82×10^{-4}	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 1-week averaged[d]	1.29×10^{-3}	NA	1.11×10^{-2}	1.11×10^{-4}	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 1-month averaged[d]	3.63×10^{-4}	NA	3.12×10^{-3}	3.12×10^{-5}	NA	NA	NA
Offsite HNO_3 concentration ($\mu g/m^3$) - 24-hour averaged[d]	9.02×10^{-2}	NA	0.777	7.77×10^{-3}	NA	NA	NA
Offsite HNO_3 concentration ($\mu g/m^3$) - annual averaged[d]	6.12×10^{-3}	NA	5.27×10^{-2}	5.27×10^{-4}	NA	NA	NA
Onsite CO concentration (mg/m^3) - 8-hour averaged[d]	0	NA	0	0	NA	NA	NA
Onsite NO_x concentration (mg/m^3) - 1-hour averaged[d]	2.07×10^{-2}	NA	0.179	1.79×10^{-3}	NA	NA	NA
Onsite SO_2 concentration (mg/m^3) - 8-hour averaged[d]	0	NA	0	0	NA	NA	NA
Onsite HNO_3 concentration (mg/m^3) - 8-hour averaged[d]	1.42×10^{-3}	NA	1.22×10^{-2}	1.22×10^{-4}	NA	NA	NA
Onsite CO_2 concentration (mg/m^3) - 8-hour averaged[d]	0	NA	0	0	NA	NA	NA
Average number of radiation workers	175	NA	409	4	NA	NA	NA
Collective worker dose (person-rem/year)	7	NA	64	0.64	NA	NA	NA
Water usage (millions of liters/year)	698	NA	1,499	15	NA	NA	NA
Electricity usage (megawatt-hour/year)	12,902	NA	27,722	277	NA	NA	NA
Steam usage (millions of kilograms/year)	63	NA	136	1	NA	NA	NA
Fuel usage (thousands of liters/year)	412	NA	886	9	NA	NA	NA
High-level liquid waste generation (millions of liters per year)	0.13	NA	1.4	0	NA	NA	NA
Equivalent DWPF canisters (per year)	3	NA	29	0	NA	NA	NA
Saltstone generation (cubic meters/year)	344	NA	3,888	0	NA	NA	NA
TRU waste generation (cubic meters/year)	0	NA	0	0	NA	NA	NA
Hazardous/mixed waste generation (cubic meters/year)	0	NA	0	0	NA	NA	NA
Low-level waste generation (cubic meters/year)	645	NA	1,276	6	NA	NA	NA

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-10. Data for impact analysis of processing and storage for vitrification (DWPF) of americium/curium solutions.[a]

Factor	Existing storage	Analyze/pretreat	Process	Interim storage[b]	Final stabilization	Storage preparation	Long-term storage[c]
Atmospheric MEI dose (rem/year)	1.5×10^{-6}	NA	8.5×10^{-6}	-	1.6×10^{-4}	NA	-
Liquid MEI dose (rem/year)	1.2×10^{-7}	NA	2.5×10^{-7}	-	2.4×10^{-5}	NA	-
Total MEI dose (rem/year)	1.6×10^{-6}	NA	8.8×10^{-6}	-	1.8×10^{-4}	NA	-
Atmospheric population dose (person-rem/year)[d]	6.7×10^{-2}	NA	3.8×10^{-1}	-	6.9	NA	-
Liquid population dose (person-rem/year)[d]	1.9310^{-3}	NA	1.46×10^{-3}	-	1.38×10^{-1}	NA	-
Total population dose (person-rem/year)	6.9×10^{-2}	NA	3.8×10^{-1}	-	7	NA	-
Offsite CO concentration (µg/m^3) - 1-hour average[e]	0	NA	0	-	0	NA	-
Offsite CO concentration (µg/m^3) - 8-hour average[e]	0	NA	0	-	0	NA	-
Offsite NO$_x$ concentration (µg/m^3) - annual average[e]	3.28×10^{-2}	NA	2.27×10^{-1}	-	0	NA	-
Offsite SO$_2$ concentration (µg/m^3) - 3-hour average[e]	0	NA	0	-	0	NA	-
Offsite SO$_2$ concentration (µg/m^3) - 24-hour average[e]	0	NA	0	-	0	NA	-
Offsite SO$_2$ concentration (µg/m^3) - annual average[e]	0	NA	0	-	0	NA	-
Offsite gaseous fluorides (µg/m^3) 12-hour average[e]	6.08×10^{-3}	NA	4.19×10^{-2}	-	0	NA	-
Offsite gaseous fluorides (µg/m^3) 24-hour average[e]	3.27×10^{-3}	NA	2.26×10^{-2}	-	0	NA	-
Offsite gaseous fluorides (µg/m^3) 1-week average[e]	1.29×10^{-3}	NA	8.87×10^{-3}	-	0	NA	-
Offsite gaseous fluorides (µg/m^3) 1-month average[e]	3.63×10^{-4}	NA	2.50×10^{-3}	-	0	NA	-
Offsite HNO$_3$ concentration (µg/m^3) - 24-hour average[e]	9.02×10^{-2}	NA	6.22×10^{-1}	-	0	NA	-
Offsite HNO$_3$ concentration (µg/m^3) - annual average[e]	6.12×10^{-3}	NA	4.22×10^{-2}	-	0	NA	-
Onsite CO concentration (mg/m^3) - 8-hour average[e]	0	NA	0	-	0	NA	-
Onsite NO$_x$ concentration (mg/m^3) - 1-hour average[e]	2.07×10^{-2}	NA	1.43×10^{-1}	-	0	NA	-
Onsite SO$_2$ concentration (mg/m^3) - 8-hour average[e]	0	NA	0	-	0	NA	-
Onsite HNO$_3$ concentration (mg/m^3) - 8-hour average[e]	1.42×10^{-3}	NA	9.77×10^{-3}	-	0	NA	-
Onsite CO$_2$ concentration (mg/m^3) - 8-hour average[e]	0	NA	0	-	0	NA	-
Average number of radiation workers	175	NA	327	-	1,500	NA	-
Collective worker dose (person-rem/year)	7	NA	51	-	60	NA	-
Water usage (millions of liters/year)	698	NA	1,199	-	1,476	NA	-
Electricity usage (megawatt-hour/year)	12,902	NA	22,176	-	50,371	NA	-
Steam usage (millions of kilograms/year)	63	NA	109	-	167	NA	-
Fuel usage (thousands of liters/year)	412	NA	709	-	0	NA	-
High-level liquid waste generation (millions of liters per year)	0.13	NA	1.4	-	0	NA	-
Equivalent DWPF canisters (per year)	3	NA	29	-	405	NA	-
Saltstone generation (cubic meters/year)	344	NA	3,888	-	0	NA	-
TRU waste generation (cubic meters/year)	0	NA	0	-	1,136	NA	-
Hazardous/mixed waste generation (cubic meters/year)	0	NA	0	-	3,055	NA	-
Low-level waste generation (cubic meters/year)	645	NA	1,160	-	2,731	NA	-

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a. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO3 = nitric acid; MEI - maximally exposed individual; NA = not applicable; NOx = nitrogen oxides, SO2 = sulfur dioxide; TRU = transuranic.

b. Atmospheric releases from the waste tank farms in 1993 contributed less than 1×10^{-8} rem to the maximally exposed individual and less than 50 person-rem to the worker collective dose. At this time, DOE does not expect any discernible increase to these or other environmental factors as a result of this phase in the stabilization alternative.

c. DOE does not expect any discernible incremental impacts as a result of this phase in the stabilization alternative.

d. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

e. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-11. Data for impact analysis of continuing storage (No Action) of americium/curium solutions.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	1.5×10^{-6}	NA	NA	NA	NA	NA	NA
Liquid MEI dose (rem/year)	1.2×10^{-7}	NA	NA	NA	NA	NA	NA
Total MEI dose (rem/year)	1.6×10^{-6}	NA	NA	NA	NA	NA	NA
Atmospheric population dose (person-rem/year)[c]	6.7×10^{-2}	NA	NA	NA	NA	NA	NA
Liquid population dose (person-rem/year)[c]	1.9×10^{-3}	NA	NA	NA	NA	NA	NA
Total population dose (person-rem/year)	6.9×10^{-2}	NA	NA	NA	NA	NA	NA
Offsite CO concentration ($\mu g/m^3$) - 1-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite CO concentration ($\mu g/m^3$) - 8-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite NO_x concentration ($\mu g/m^3$) - annual average[d]	3.28×10^{-2}	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration ($\mu g/m^3$) - 3-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration ($\mu g/m^3$) - 24-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration ($\mu g/m^3$) - annual average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 12-hour average[d]	6.08×10^{-3}	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 24-hour average[d]	3.27×10^{-3}	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 1-week average[d]	1.29×10^{-3}	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 1-month average[d]	3.63×10^{-4}	NA	NA	NA	NA	NA	NA
Offsite HNO_3 concentration ($\mu g/m^3$) - 24-hour average[d]	9.02×10^{-2}	NA	NA	NA	NA	NA	NA
Offsite HNO_3 concentration ($\mu g/m^3$) - annual average[d]	6.12×10^{-3}	NA	NA	NA	NA	NA	NA
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	NA	NA	NA	NA
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	2.07×10^{-2}	NA	NA	NA	NA	NA	NA
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	NA	NA	NA	NA
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	1.42×10^{-3}	NA	NA	NA	NA	NA	NA
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	NA	NA	NA	NA
Average number of radiation workers	175	NA	NA	NA	NA	NA	NA
Collective worker dose (person-rem/year)	7	NA	NA	NA	NA	NA	NA
Water usage (millions of liters/year)	698	NA	NA	NA	NA	NA	NA
Electricity usage (megawatt-hour/year)	12,902	NA	NA	NA	NA	NA	NA
Steam usage (millions of kilograms/year)	63	NA	NA	NA	NA	NA	NA
Fuel usage (thousands of liters/year)	412	NA	NA	NA	NA	NA	NA
High-level liquid waste generation (millions of liters per year)	0.13	NA	NA	NA	NA	NA	NA
Equivalent DWPF canisters (per year)	3	NA	NA	NA	NA	NA	NA
Saltstone generation (cubic meters/year)	344	NA	NA	NA	NA	NA	NA
TRU waste generation (cubic meters/year)	0	NA	NA	NA	NA	NA	NA
Hazardous/mixed waste generation (cubic meters/year)	0	NA	NA	NA	NA	NA	NA
Low-level waste generation (cubic meters/year)	645	NA	NA	NA	NA	NA	NA

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-12. Data for impact analysis of vitrification (F-Canyon) of americium/curium targets.[a,b]

Factor	Existing storage	Analyze/ pretreat	Process	Interim storage	Final stabilization	Storage preparation	Long-term storage
Atmospheric MEI dose (rem/year)	1.4×10^{-10}	NA	1.1×10^{-5}	1.1×10^{-7}	NA	NA	NA
Liquid MEI dose (rem/year)	3.5×10^{-7}	NA	7.4×10^{-7}	3.9×10^{-9}	NA	NA	NA
Total MEI dose (rem/year)	3.5×10^{-7}	NA	1.1×10^{-5}	1.1×10^{-7}	NA	NA	NA
Atmospheric population dose (person-rem/year)[c]	5.8×10^{-6}	NA	0.48	4.8×10^{-3}	NA	NA	NA
Liquid population dose (person-rem/year)[c]	1.0×10^{-3}	NA	3.3×10^{-3}	6.1×10^{-5}	NA	NA	NA
Total population dose (person-rem/year)	1.0×10^{-3}	NA	0.48	4.8×10^{-3}	NA	NA	NA
Offsite CO concentration ($\mu g/m^3$) - 1-hour average[d]	0	NA	0	0	NA	NA	NA
Offsite CO concentration ($\mu g/m^3$) - 8-hour average[d]	0	NA	0	0	NA	NA	NA
Offsite NO_x concentration ($\mu g/m^3$) - annual average[d]	0	NA	0.283	2.83×10^{-3}	NA	NA	NA
Offsite SO_2 concentration ($\mu g/m^3$) - 3-hour average[d]	0	NA	0	0	NA	NA	NA
Offsite SO_2 concentration ($\mu g/m^3$) - 24-hour average[d]	0	NA	0	0	NA	NA	NA
Offsite SO_2 concentration ($\mu g/m^3$) - annual average[d]	0	NA	0	0	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 12-hour average[d]	0	NA	5.24×10^{-2}	5.24×10^{-4}	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 24-hour average[d]	0	NA	2.82×10^{-2}	2.82×10^{-4}	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 1-week average[d]	0	NA	1.11×10^{-2}	1.11×10^{-4}	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 1-month average[d]	0	NA	3.12×10^{-3}	3.12×10^{-5}	NA	NA	NA
Offsite HNO_3 concentration ($\mu g/m^3$) - 24-hour average[d]	0	NA	0.777	7.77×10^{-3}	NA	NA	NA
Offsite HNO_3 concentration ($\mu g/m^3$) - annual average[d]	0	NA	5.27×10^{-2}	5.27×10^{-4}	NA	NA	NA
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	NA	NA	NA
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	0	NA	0.179	1.79×10^{-3}	NA	NA	NA
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	NA	NA	NA
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	0	NA	1.22×10^{-2}	1.22×10^{-4}	NA	NA	NA
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	NA	NA	NA
Average number of radiation workers	34	NA	477	4	NA	NA	NA
Collective worker dose (person-rem/year)	1.4	NA	67	0.64	NA	NA	NA
Water usage (millions of liters/year)	28	NA	1,534	15	NA	NA	NA
Electricity usage (megawatt-hour/year)	1,144	NA	29,150	277	NA	NA	NA
Steam usage (millions of kilograms/year)	9	NA	147	1	NA	NA	NA
Fuel usage (thousands of liters/year)	6	NA	893	9	NA	NA	NA
High-level liquid waste generation (millions of liters per year)	0.041	NA	1.5	0	NA	NA	NA
Equivalent DWPF canisters (per year)	0	NA	29	0	NA	NA	NA
Saltstone generation (cubic meters/year)	11	NA	3,898	0	NA	NA	NA
TRU waste generation (cubic meters/year)	0	NA	0	0	NA	NA	NA
Hazardous/mixed waste generation (cubic meters/year)	0	NA	0	0	NA	NA	NA
Low-level waste generation (cubic meters/year)	128	NA	1,403	6	NA	NA	NA

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

Table D-13. Data for impact analysis of processing americium/curium targets to oxide.[a,b]

Factor	Existing storage	Analyze/ pretreat	Process	Interim storage	Final stabilization	Storage preparation	Long-term storage
Atmospheric MEI dose (rem/year)	1.4×10^{-10}	NA	1.1×10^{-5}	1.1×10^{-7}	NA	NA	NA
Liquid MEI dose (rem/year)	3.5×10^{-7}	NA	7.4×10^{-7}	3.9×10^{-9}	NA	NA	NA
Total MEI dose (rem/year)	3.5×10^{-7}	NA	1.2×10^{-5}	1.1×10^{-7}	NA	NA	NA
Atmospheric population dose (person-rem/year)[c]	5.8×10^{-6}	NA	0.48	4.8×10^{-3}	NA	NA	NA
Liquid population dose (person-rem/year)[c]	1.0×10^{-3}	NA	3.3×10^{-3}	6.1×10^{-5}	NA	NA	NA
Total population dose (person-rem/year)	1.0×10^{-3}	NA	0.48	4.8×10^{-3}	NA	NA	NA
Offsite CO concentration (μg/m^3) - 1-hour average[d]	0	NA	0	0	NA	NA	NA
Offsite CO concentration (μg/m^3) - 8-hour average[d]	0	NA	0	0	NA	NA	NA
Offsite NO_x concentration (μg/m^3) - annual average[d]	0.0328	NA	0.283	2.83×10^{-3}	NA	NA	NA
Offsite SO_2 concentration (μg/m^3) - 3-hour average[d]	0	NA	0	0	NA	NA	NA
Offsite SO_2 concentration (μg/m^3) - 24-hour average[d]	0	NA	0	0	NA	NA	NA
Offsite SO_2 concentration (μg/m^3) - annual average[d]	0	NA	0	0	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 12-hour average[d]	6.08×10^{-3}	NA	5.24×10^{-2}	5.24×10^{-4}	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 24-hour average[d]	3.27×10^{-3}	NA	2.82×10^{-2}	2.82×10^{-4}	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 1-week average[d]	1.29×10^{-3}	NA	1.11×10^{-2}	1.11×10^{-4}	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 1-month average[d]	3.63×10^{-4}	NA	3.12×10^{-3}	3.12×10^{-5}	NA	NA	NA
Offsite HNO_3 concentration (μg/m^3) - 24-hour average[d]	9.02×10^{-2}	NA	0.777	7.77×10^{-3}	NA	NA	NA
Offsite HNO_3 concentration (μg/m^3) - annual average[d]	6.12×10^{-3}	NA	5.27×10^{-2}	5.27×10^{-4}	NA	NA	NA
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	NA	NA	NA
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	0	NA	0.179	1.79×10^{-3}	NA	NA	NA
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	NA	NA	NA
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	0	NA	1.22×10^{-2}	1.22×10^{-4}	NA	NA	NA
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	NA	NA	NA
Average number of radiation workers	34	NA	477	4	NA	NA	NA
Collective worker dose (person-rem/year)	1.4	NA	67	0.64	NA	NA	NA
Water usage (millions of liters/year)	28	NA	1,534	15	NA	NA	NA
Electricity usage (megawatt-hour/year)	1,144	NA	29,150	277	NA	NA	NA
Steam usage (millions of kilograms/year)	9	NA	147	1	NA	NA	NA
Fuel usage (thousands of liters/year)	6	NA	893	9	NA	NA	NA
High-level liquid waste generation (millions of liters per year)	0.041	NA	1.5	0	NA	NA	NA
Equivalent DWPF canisters (per year)	0	NA	29	0	NA	NA	NA
Saltstone generation (cubic meters/year)	11	NA	3,898	0	NA	NA	NA
TRU waste generation (cubic meters/year)	0	NA	0	0	NA	NA	NA
Hazardous/mixed waste generation (cubic meters/year)	0	NA	0	0	NA	NA	NA
Low-level waste generation (cubic meters/year)	128	NA	1,403	6	NA	NA	NA

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-14. Data for impact analysis of continuing storage (No Action) of americium/curium targets.[a,b]

Factor	Existing storage	Analyze/ pretreat	Process	Interim storage	Final stabilization	Storage preparation	Long-term storage
Atmospheric MEI dose (rem/year)	1.4×10^{-10}	NA	NA	NA	NA	NA	NA
Liquid MEI dose (rem/year)	3.5×10^{-7}	NA	NA	NA	NA	NA	NA
Total MEI dose (rem/year)	3.5×10^{-7}	NA	NA	NA	NA	NA	NA
Atmospheric population dose (person-rem/year)[c]	5.8×10^{-6}	NA	NA	NA	NA	NA	NA
Liquid population dose (person-rem/year)[c]	1.0×10^{-3}	NA	NA	NA	NA	NA	NA
Total population dose (person-rem/year)	1.0×10^{-3}	NA	NA	NA	NA	NA	NA
Offsite CO concentration ($\mu g/m^3$) - 1-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite CO concentration ($\mu g/m^3$) - 8-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite NO_x concentration ($\mu g/m^3$) - annual average[d]	0	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration ($\mu g/m^3$) - 3-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration ($\mu g/m^3$) - 24-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration ($\mu g/m^3$) - annual average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 12-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 24-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 1-week average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 1-month average[d]	0	NA	NA	NA	NA	NA	NA
Offsite HNO_3 concentration ($\mu g/m^3$) - 24-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite HNO_3 concentration ($\mu g/m^3$) - annual average[d]	0	NA	NA	NA	NA	NA	NA
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	NA	NA	NA	NA
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	0	NA	NA	NA	NA	NA	NA
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	NA	NA	NA	NA
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	NA	NA	NA	NA
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	NA	NA	NA	NA
Average number of radiation workers	34	NA	NA	NA	NA	NA	NA
Collective worker dose (person-rem/year)	1.4	NA	NA	NA	NA	NA	NA
Water usage (millions of liters/year)	28	NA	NA	NA	NA	NA	NA
Electricity usage (megawatt-hour/year)	1,144	NA	NA	NA	NA	NA	NA
Steam usage (millions of kilograms/year)	9	NA	NA	NA	NA	NA	NA
Fuel usage (thousands of liters/year)	6	NA	NA	NA	NA	NA	NA
High-level liquid waste generation (millions of liters per year)	0.041	NA	NA	NA	NA	NA	NA
Equivalent DWPF canisters (per year)	0	NA	NA	NA	NA	NA	NA
Saltstone generation (cubic meters/year)	11	NA	NA	NA	NA	NA	NA
TRU waste generation (cubic meters/year)	0	NA	NA	NA	NA	NA	NA
Hazardous/mixed waste generation (cubic meters/year)	0	NA	NA	NA	NA	NA	NA
Low-level waste generation (cubic meters/year)	128	NA	NA	NA	NA	NA	NA

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-15. Data for impact analysis of processing neptunium to oxide.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	1.0×10^{-6}	NA	8.3×10^{-4}	NA	NA	6.6×10^{-9}	4.4×10^{-9}
Liquid MEI dose (rem/year)	4.1×10^{-6}	NA	1.2×10^{-5}	NA	NA	7.3×10^{-10}	4.3×10^{-10}
Total MEI dose (rem/year)	5.1×10^{-6}	NA	8.4×10^{-4}	NA	NA	7.3×10^{-9}	4.8×10^{-9}
Atmospheric population dose (person-rem/year)[c]	4.2×10^{-2}	NA	34	NA	NA	2.9×10^{-4}	2.0×10^{-4}
Liquid population dose (person-rem/year)[c]	1.2×10^{-2}	NA	4.2×10^{-2}	NA	NA	6.4×10^{-6}	6.8×10^{-6}
Total population dose (person-rem/year)	5.4×10^{-2}	NA	34	NA	NA	3.0×10^{-4}	2.0×10^{-4}
Offsite CO concentration ($\mu g/m^3$) - 1-hour average[d]	1.78	NA	13.3	NA	NA	0	0
Offsite CO concentration ($\mu g/m^3$) - 8-hour average[d]	0.426	NA	3.18	NA	NA	0	0
Offsite NO_x concentration ($\mu g/m^3$) - annual average[d]	1.94×10^{-2}	NA	0.103	NA	NA	0	0
Offsite SO_2 concentration ($\mu g/m^3$) - 3-hour average[d]	1.05×10^{-3}	NA	7.75×10^{-3}	NA	NA	0	0
Offsite SO_2 concentration ($\mu g/m^3$) - 24-hour average[d]	2.35×10^{-4}	NA	1.74×10^{-3}	NA	NA	0	0
Offsite SO_2 concentration ($\mu g/m^3$) - annual average[d]	1.48×10^{-5}	NA	1.10×10^{-4}	NA	NA	0	0
Offsite gaseous fluorides ($\mu g/m^3$) 12-hour average[d]	0	NA	1.50×10^{-5}	NA	NA	0	0
Offsite gaseous fluorides ($\mu g/m^3$) 24-hour average[d]	0	NA	5.95×10^{-6}	NA	NA	0	0
Offsite gaseous fluorides ($\mu g/m^3$) 1-week average[d]	0	NA	3.16×10^{-6}	NA	NA	0	0
Offsite gaseous fluorides ($\mu g/m^3$) 1-month average[d]	0	NA	8.92×10^{-7}	NA	NA	0	0
Offsite HNO_3 concentration ($\mu g/m^3$) - 24-hour average[d]	5.20×10^{-3}	NA	2.76×10^{-2}	NA	NA	0	0
Offsite HNO_3 concentration ($\mu g/m^3$) - annual average[d]	3.26×10^{-4}	NA	1.74×10^{-3}	NA	NA	0	0
Onsite CO concentration (mg/m^3) - 8-hour average[d]	2.84×10^{-3}	NA	2.12×10^{-2}	NA	NA	0	0
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	1.02×10^{-2}	NA	5.45×10^{-2}	NA	NA	0	0
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	4.15×10^{-6}	NA	3.08×10^{-5}	NA	NA	0	0
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	9.17×10^{-5}	NA	4.88×10^{-4}	NA	NA	0	0
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	2.05×10^{-6}	NA	1.52×10^{-5}	NA	NA	0	0
Average number of radiation workers	152	NA	508	NA	NA	157	80
Collective worker dose (person-rem/year)	1.52	NA	49	NA	NA	31	16
Water usage (millions of liters/year)	185	NA	347	NA	NA	500	454
Electricity usage (megawatt-hour/year)	14,157	NA	29,887	NA	NA	6,620	6,018
Steam usage (millions of kilograms/year)	84	NA	145	NA	NA	7	6
Fuel usage (thousands of liters/year)	531	NA	907	NA	NA	21	19
High-level liquid waste generation (millions of liters per year)	0.13	NA	2.0	NA	NA	0	0
Equivalent DWPF canisters (per year)	2	NA	15	NA	NA	0	0
Saltstone generation (cubic meters/year)	356	NA	5,570	NA	NA	0	0
TRU waste generation (cubic meters/year)	0	NA	98	NA	NA	1	0
Hazardous/mixed waste generation (cubic meters/year)	0	NA	119	NA	NA	1	0
Low-level waste generation (cubic meters/year)	570	NA	1,799	NA	NA	7	5

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-16. Data for impact analysis of vitrification (F-Canyon) of neptunium.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	1.0×10^{-6}	NA	2.9×10^{-6}	NA	2.1×10^{-6}	NA	4.4×10^{-9}
Liquid MEI dose (rem/year)	4.1×10^{-6}	NA	4.3×10^{-6}	NA	6.3×10^{-8}	NA	4.3×10^{-10}
Total MEI dose (rem/year)	5.1×10^{-6}	NA	7.1×10^{-6}	NA	2.2×10^{-6}	NA	4.8×10^{-9}
Atmospheric population dose (person-rem/year)[c]	4.2×10^{-2}	NA	0.13	NA	9.5×10^{-2}	NA	2.0×10^{-4}
Liquid population dose (person-rem/year)[c]	1.2×10^{-2}	NA	1.5×10^{-2}	NA	3.7×10^{-4}	NA	6.8×10^{-6}
Total population dose (person-rem/year)	5.4×10^{-2}	NA	0.14	NA	9.5×10^{-2}	NA	2.0×10^{-4}
Offsite CO concentration (μg/m^3) - 1-hour average[d]	1.78	NA	13.3	NA	0	NA	0
Offsite CO concentration (μg/m^3) - 8-hour average[d]	0.426	NA	3.18	NA	0	NA	0
Offsite NO_x concentration (μg/m^3) - annual average[d]	1.94×10^{-2}	NA	0.103	NA	5.66×10^{-2}	NA	0
Offsite SO_2 concentration (μg/m^3) - 3-hour average[d]	1.05×10^{-3}	NA	7.75×10^{-3}	NA	0	NA	0
Offsite SO_2 concentration (μg/m^3) - 24-hour average[d]	2.35×10^{-4}	NA	1.75×10^{-3}	NA	0	NA	0
Offsite SO_2 concentration (μg/m^3) - annual average[d]	1.48×10^{-5}	NA	1.10×10^{-4}	NA	0	NA	0
Offsite gaseous fluorides (μg/m^3) 12-hour average[d]	0	NA	1.50×10^{-5}	NA	1.05×10^{-2}	NA	0
Offsite gaseous fluorides (μg/m^3) 24-hour average[d]	0	NA	5.95×10^{-6}	NA	5.65×10^{-3}	NA	0
Offsite gaseous fluorides (μg/m^3) 1-week average[d]	0	NA	3.16×10^{-6}	NA	2.22×10^{-3}	NA	0
Offsite gaseous fluorides (μg/m^3) 1-month average[d]	0	NA	8.92×10^{-3}	NA	6.25×10^{-4}	NA	0
Offsite HNO_3 concentration (μg/m^3) - 24-hour average[d]	5.20×10^{-3}	NA	0.276	NA	0.155	NA	0
Offsite HNO_3 concentration (μg/m^3) - annual average[d]	3.26×10^{-4}	NA	1.74×10^{-3}	NA	1.05×10^{-2}	NA	0
Onsite CO concentration (mg/m^3) - 8-hour average[d]	2.84×10^{-3}	NA	2.12×10^{-2}	NA	0	NA	0
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	1.02×10^{-2}	NA	5.45×10^{-2}	NA	3.58×10^{-2}	NA	0
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	4.15×10^{-6}	NA	3.08×10^{-5}	NA	0	NA	0
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	9.17×10^{-5}	NA	4.88×10^{-4}	NA	2.44×10^{-3}	NA	0
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	2.05×10^{-6}	NA	1.52×10^{-5}	NA	0	NA	0
Average number of radiation workers	152	NA	359	NA	82	NA	80
Collective worker dose (person-rem/year)	1.52	NA	14	NA	13	NA	16
Water usage (millions of liters/year)	185	NA	1,101	NA	300	NA	454
Electricity usage (megawatt-hour/year)	14,157	NA	31,091	NA	5,544	NA	6,018
Steam usage (millions of kilograms/year)	84	NA	167	NA	27	NA	6
Fuel usage (thousands of liters/year)	531	NA	1,073	NA	177	NA	19
High-level liquid waste generation (millions of liters per year)	0.13	NA	0.31	NA	0	NA	3.8×10^{-5}
Equivalent DWPF canisters (per year)	2	NA	6	NA	0	NA	0
Saltstone generation (cubic meters/year)	356	NA	856	NA	0	NA	0
TRU waste generation (cubic meters/year)	0	NA	0	NA	0	NA	0
Hazardous/mixed waste generation (cubic meters/year)	0	NA	0	NA	0	NA	0
Low-level waste generation (cubic meters/year)	570	NA	1,469	NA	116	NA	5

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

Table D-17. Data for impact analysis of processing and storage for vitrification (DWPF) of neptunium.[a,b]

Factor	Existing storage	Analyze/pretreat	Process	Interim storage[c]	Final stabilization	Storage preparation	Long-term storage[d]
Atmospheric MEI dose (rem/year)	1.0×10^{-6}	NA	6.6×10^{-4}	-	1.6×10^{-4}	NA	-
Liquid MEI dose (rem/year)	4.1×10^{-6}	NA	9.8×10^{-6}	-	2.4×10^{-5}	NA	-
Total MEI dose (rem/year)	5.1×10^{-6}	NA	6.7×10^{-4}	-	1.8×10^{-4}	NA	-
Atmospheric population dose (person-rem/year)[e]	4.2×10^{-2}	NA	27	-	6.9	NA	-
Liquid population dose (person-rem/year)[e]	1.2×10^{-2}	NA	3.3×10^{-2}	-	1.4×10^{-1}	NA	-
Total population dose (person-rem/year)	5.4×10^{-2}	NA	27	-	7	NA	-
Offsite CO concentration (μg/m^3) - 1-hour average[f]	1.78	NA	10.6	-	0	NA	-
Offsite CO concentration (μg/m^3) - 8-hour average[f]	4.26×10^{-1}	NA	2.54	-	0	NA	-
Offsite NO_x concentration (μg/m^3) - annual average[f]	1.94×10^{-2}	NA	8.25×10^{-2}	-	0	NA	-
Offsite SO_2 concentration (μg/m^3) - 3-hour average[f]	1.05×10^{-3}	NA	6.21×10^{-3}	-	0	NA	-
Offsite SO_2 concentration (μg/m^3) - 24-hour average[f]	2.35×10^{-4}	NA	1.40×10^{-3}	-	0	NA	-
Offsite SO_2 concentration (μg/m^3) - annual average[f]	1.48×10^{-5}	NA	8.77×10^{-5}	-	0	NA	-
Offsite gaseous fluorides (μg/m^3) 12-hour average[f]	0	NA	7.48×10^{-6}	-	0	NA	-
Offsite gaseous fluorides (μg/m^3) 24-hour average[f]	0	NA	2.98×10^{-6}	-	0	NA	-
Offsite gaseous fluorides (μg/m^3) 1-week average[f]	0	NA	1.58×10^{-6}	-	0	NA	-
Offsite gaseous fluorides (μg/m^3) 1-month average[f]	0	NA	4.46×10^{-7}	-	0	NA	-
Offsite HNO_3 concentration (μg/m^3) - 24-hour average[f]	5.20×10^{-3}	NA	2.21×10^{-2}	-	0	NA	-
Offsite HNO_3 concentration (μg/m^3) - annual average[f]	3.26×10^{-4}	NA	1.39×10^{-3}	-	0	NA	-
Onsite CO concentration (mg/m^3) - 8-hour average[f]	2.84×10^{-3}	NA	1.70×10^{-2}	-	0	NA	-
Onsite NO_x concentration (mg/m^3) - 1-hour average[f]	1.02×10^{-2}	NA	4.36×10^{-2}	-	0	NA	-
Onsite SO_2 concentration (mg/m^3) - 8-hour average[f]	4.15×10^{-6}	NA	2.46×10^{-5}	-	0	NA	-
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[f]	9.17×10^{-5}	NA	3.90×10^{-4}	-	0	NA	-
Onsite CO_2 concentration (mg/m^3) - 8-hour average[f]	2.05×10^{-6}	NA	1.22×10^{-5}	-	0	NA	-
Average number of radiation workers	152	NA	351	-	1,500	NA	-
Collective worker dose (person-rem/year)	1.52	NA	14	-	60	NA	-
Water usage (millions of liters/year)	185	NA	308	-	1,476	NA	-
Electricity usage (megawatt-hour/year)	14,157	NA	23,595	-	50,371	NA	-
Steam usage (millions of kilograms/year)	84	NA	140	-	167	NA	-
Fuel usage (thousands of liters/year)	531	NA	886	-	0	NA	-
High-level liquid waste generation (millions of liters per year)	0.13	NA	2.0	-	0	NA	-
Equivalent DWPF canisters (per year)	2	NA	15	-	405	NA	-
Saltstone generation (cubic meters/year)	356	NA	5,570	-	0	NA	-
TRU waste generation (cubic meters/year)	0	NA	0	-	1,136	NA	-
Hazardous/mixed waste generation (cubic meters/year)	0	NA	0	-	3,055	NA	-
Low-level waste generation (cubic meters/year)	570	NA	1,029	-	2,731	NA	-

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Atmospheric releases from the waste tank farms in 1993 contributed less than 1×10^{-8} rem to the maximally exposed individual and less than 50 person-rem to the worker collective dose. At this time, DOE does not expect any discernible increase to these or other environmental factors as a result of this phase in the stabilization alternative.

d. DOE does not expect any discernible incremental impacts as a result of this phase in the stabilization alternative.

e. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

f. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-18. Data for impact analysis of continuing storage (No Action) of neptunium.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	1.0×10^{-6}	NA	NA	NA	NA	NA	NA
Liquid MEI dose (rem/year)	4.1×10^{-6}	NA	NA	NA	NA	NA	NA
Total MEI dose (rem/year)	5.1×10^{-6}	NA	NA	NA	NA	NA	NA
Atmospheric population dose (person-rem/year)[c]	4.2×10^{-2}	NA	NA	NA	NA	NA	NA
Liquid population dose (person-rem/year)[c]	1.2×10^{-2}	NA	NA	NA	NA	NA	NA
Total population dose (person-rem/year)	5.4×10^{-2}	NA	NA	NA	NA	NA	NA
Offsite CO concentration (μg/m^3) - 1-hour average[d]	1.78	NA	NA	NA	NA	NA	NA
Offsite CO concentration (μg/m^3) - 8-hour average[d]	0.426	NA	NA	NA	NA	NA	NA
Offsite NO_x concentration (μg/m^3) - annual average[d]	1.94×10^{-2}	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration (μg/m^3) - 3-hour average[d]	1.05×10^{-3}	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration (μg/m^3) - 24-hour average[d]	2.35×10^{-4}	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration (μg/m^3) - annual average[d]	1.48×10^{-5}	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 12-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 24-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 1-week average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 1-month average[d]	0	NA	NA	NA	NA	NA	NA
Offsite HNO_3 concentration (μg/m^3) - 24-hour average[d]	5.20×10^{-3}	NA	NA	NA	NA	NA	NA
Offsite HNO_3 concentration (μg/m^3) - annual average[d]	3.26×10^{-4}	NA	NA	NA	NA	NA	NA
Onsite CO concentration (mg/m^3) - 8-hour average[d]	2.84×10^{-3}	NA	NA	NA	NA	NA	NA
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	1.02×10^{-2}	NA	NA	NA	NA	NA	NA
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	4.15×10^{-6}	NA	NA	NA	NA	NA	NA
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	9.17×10^{-5}	NA	NA	NA	NA	NA	NA
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	2.05×10^{-6}	NA	NA	NA	NA	NA	NA
Average number of radiation workers	152	NA	NA	NA	NA	NA	NA
Collective worker dose (person-rem/year)	1.52	NA	NA	NA	NA	NA	NA
Water usage (millions of liters/year)	185	NA	NA	NA	NA	NA	NA
Electricity usage (megawatt-hour/year)	14,157	NA	NA	NA	NA	NA	NA
Steam usage (millions of kilograms/year)	84	NA	NA	NA	NA	NA	NA
Fuel usage (thousands of liters/year)	531	NA	NA	NA	NA	NA	NA
High-level liquid waste generation (millions of liters per year)	0.13	NA	NA	NA	NA	NA	NA
Equivalent DWPF canisters (per year)	2	NA	NA	NA	NA	NA	NA
Saltstone generation (cubic meters/year)	356	NA	NA	NA	NA	NA	NA
TRU waste generation (cubic meters/year)	0	NA	NA	NA	NA	NA	NA
Hazardous/mixed waste generation (cubic meters/year)	0	NA	NA	NA	NA	NA	NA
Low-level waste generation (cubic meters/year)	570	NA	NA	NA	NA	NA	NA

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

TC

Table D-19. Data for impact analysis of processing H-Canyon plutonium-239 solutions to oxide.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	9.4×10^{-7}	NA	1.7×10^{-4}	2.3×10^{-9}	NA	6.6×10^{-9}	4.4×10^{-9}
Liquid MEI dose (rem/year)	3.7×10^{-6}	NA	6.2×10^{-6}	2.3×10^{-10}	NA	7.3×10^{-10}	4.3×10^{-10}
Total MEI dose (rem/year)	4.7×10^{-6}	NA	1.7×10^{-4}	2.6×10^{-9}	NA	7.3×10^{-9}	4.8×10^{-9}
Atmospheric population dose (person-rem/year)[c]	3.8×10^{-2}	NA	6.8	1.0×10^{-4}	NA	2.9×10^{-4}	2.0×10^{-4}
Liquid population dose (person-rem/year)[c]	1.1×10^{-2}	NA	1.9×10^{-2}	3.6×10^{-6}	NA	6.4×10^{-6}	6.8×10^{-6}
Total population dose (person-rem/year)	4.9×10^{-2}	NA	6.8	1.1×10^{-4}	NA	3.0×10^{-4}	2.0×10^{-4}
Offsite CO concentration (μg/m^3) - 1-hour average[d]	1.30	NA	3.96	0	NA	0	0
Offsite CO concentration (μg/m^3) - 8-hour average[d]	0.311	NA	0.947	0	NA	0	0
Offsite NO_x concentration (μg/m^3) - annual average[d]	1.20×10^{-2}	NA	3.26×10^{-2}	0	NA	0	0
Offsite SO_2 concentration (μg/m^3) - 3-hour average[d]	7.54×10^{-4}	NA	2.32×10^{-3}	0	NA	0	0
Offsite SO_2 concentration (μg/m^3) - 24-hour average[d]	1.70×10^{-4}	NA	5.21×10^{-4}	0	NA	0	0
Offsite SO_2 concentration (μg/m^3) - annual average[d]	1.07×10^{-5}	NA	3.27×10^{-5}	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 12-hour average[d]	0	NA	0	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 24-hour average[d]	0	NA	0	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 1-week average[d]	0	NA	0	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 1-month average[d]	0	NA	0	0	NA	0	0
Offsite HNO_3 concentration (μg/m^3) - 24-hour average[d]	3.21×10^{-3}	NA	8.74×10^{-3}	0	NA	0	0
Offsite HNO_3 concentration (μg/m^3) - annual average[d]	2.01×10^{-4}	NA	5.49×10^{-4}	0	NA	0	0
Onsite CO concentration (mg/m^3) - 8-hour average[d]	2.08×10^{-3}	NA	6.32×10^{-3}	0	NA	0	0
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	6.32×10^{-3}	NA	1.72×10^{-2}	0	NA	0	0
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	2.99×10^{-6}	NA	9.19×10^{-6}	0	NA	0	0
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	5.65×10^{-5}	NA	1.54×10^{-4}	0	NA	0	0
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	1.47×10^{-6}	NA	4.57×10^{-6}	0	NA	0	0
Average number of radiation workers	134	NA	291	48	NA	157	80
Collective worker dose (person-rem/year)	1.34	NA	36	2	NA	31	16
Water usage (millions of liters/year)	174	NA	212	283	NA	500	454
Electricity usage (megawatt-hour/year)	13,299	NA	19,591	3,755	NA	6,619	6,018
Steam usage (millions of kilograms/year)	79	NA	84	4	NA	7	6
Fuel usage (thousands of liters/year)	499	NA	520	12	NA	21	19
High-level liquid waste generation (millions of liters per year)	0.12	NA	0.12	0	NA	0	0
Equivalent DWPF canisters (per year)	2	NA	2	0	NA	0	0
Saltstone generation (cubic meters/year)	325	NA	325	0	NA	0	0
TRU waste generation (cubic meters/year)	0	NA	98	2	NA	1	0
Hazardous/mixed waste generation (cubic meters/year)	0	NA	119	1	NA	1	0
Low-level waste generation (cubic meters/year)	556	NA	1,326	972	NA	7	5

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

TC

Table D-20. Data for impact analysis of processing and storage for vitrification (DWPF) of H-Canyon plutonium-239 solutions.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage[c]	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage[d]
Atmospheric MEI dose (rem/year)	9.4×10^{-7}	NA	6.6×10^{-4}	-	1.6×10^{-4}	NA	-
Liquid MEI dose (rem/year)	3.7×10^{-6}	NA	9.8×10^{-6}	-	2.4×10^{-5}	NA	-
Total MEI dose (rem/year)	4.7×10^{-6}	NA	6.7×10^{-4}	-	1.8×10^{-4}	NA	-
Atmospheric population dose (person-rem/year)[e]	3.8×10^{-2}	NA	27	-	6.9	NA	-
Liquid population dose (person-rem/year)[e]	1.1×10^{-2}	NA	3.3×10^{-2}	-	0.14	NA	-
Total population dose (person-rem/year)	4.9×10^{-2}	NA	27	-	7	NA	-
Offsite CO concentration (μg/m^3) - 1-hour average[f]	1.30	NA	10.6	-	0	NA	-
Offsite CO concentration (μg/m^3) - 8-hour average[f]	0.311	NA	2.54	-	0	NA	-
Offsite NO_x concentration (μg/m^3) - annual average[f]	1.20×10^{-2}	NA	8.25×10^{-2}	-	0	NA	-
Offsite SO_2 concentration (μg/m^3) - 3-hour average[f]	7.54×10^{-4}	NA	6.21×10^{-3}	-	0	NA	-
Offsite SO_2 concentration (μg/m^3) - 24-hour average[f]	1.70×10^{-4}	NA	1.40×10^{-3}	-	0	NA	-
Offsite SO_2 concentration (μg/m^3) - annual average[f]	1.07×10^{-5}	NA	8.77×10^{-5}	-	0	NA	-
Offsite gaseous fluorides (μg/m^3) 12-hour average[f]	0	NA	7.48×10^{-6}	-	0	NA	-
Offsite gaseous fluorides (μg/m^3) 24-hour average[f]	0	NA	2.98×10^{-6}	-	0	NA	-
Offsite gaseous fluorides (μg/m^3) 1-week average[f]	0	NA	1.58×10^{-6}	-	0	NA	-
Offsite gaseous fluorides (μg/m^3) 1-month average[f]	0	NA	4.46×10^{-7}	-	0	NA	-
Offsite HNO_3 concentration (μg/m^3) - 24-hour average[f]	3.21×10^{-3}	NA	2.21×10^{-2}	-	0	NA	-
Offsite HNO_3 concentration (μg/m^3) - annual average[f]	2.01×10^{-4}	NA	1.39×10^{-3}	-	0	NA	-
Onsite CO concentration (mg/m^3) - 8-hour average[f]	2.08×10^{-3}	NA	1.70×10^{-2}	-	0	NA	-
Onsite NO_x concentration (mg/m^3) - 1-hour average[f]	6.32×10^{-3}	NA	4.36×10^{-2}	-	0	NA	-
Onsite SO_2 concentration (mg/m^3) - 8-hour average[f]	2.99×10^{-6}	NA	2.46×10^{-5}	-	0	NA	-
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[f]	5.65×10^{-5}	NA	3.90×10^{-4}	-	0	NA	-
Onsite CO_2 concentration (mg/m^3) - 8-hour average[f]	1.47×10^{-6}	NA	1.22×10^{-5}	-	0	NA	-
Average number of radiation workers	134	NA	351	-	1,500	NA	-
Collective worker dose (person-rem/year)	1.34	NA	14	-	60	NA	-
Water usage (millions of liters/year)	174	NA	308	-	1,476	NA	-
Electricity usage (megawatt-hour/year)	13,299	NA	23,595	-	50,371	NA	-
Steam usage (millions of kilograms/year)	79	NA	140	-	167	NA	-
Fuel usage (thousands of liters/year)	499	NA	886	-	499	NA	-
High-level liquid waste generation (millions of liters/year)	0.12	NA	2.0	-	0	NA	-
Equivalent DWPF canisters (per year)	2	NA	15	-	405	NA	-
Saltstone generation (cubic meters/year)	325	NA	5,570	-	0	NA	-
TRU waste generation (cubic meters/year)	0	NA	0	-	1,136	NA	-
Hazardous/mixed waste generation (cubic meters/year)	0	NA	0	-	3,055	NA	-
Low-level waste generation (cubic meters/year)	556	NA	1,029	-	2,731	NA	-

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Atmospheric releases from the waste tank farms in 1993 contributed less than 1×10^{-8} rem to the maximally exposed individual and less than 50 person-rem to the worker collective dose. At this time, DOE does not expect any discernible increase to these or other environmental factors as a result of this phase in the stabilization alternative.

d. DOE does not expect any discernible incremental impacts as a result of this phase in the stabilization alternative.

e. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

f. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

TC

Table D-21. Data for impact analysis of vitrification (F-Canyon) of H-Canyon plutonium-239 solutions.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	9.4×10^{-7}	NA	2.8×10^{-6}	NA	2.1×10^{-6}	NA	4.4×10^{-9}
Liquid MEI dose (rem/year)	3.7×10^{-6}	NA	3.9×10^{-6}	NA	6.3×10^{-8}	NA	4.3×10^{-10}
Total MEI dose (rem/year)	4.7×10^{-6}	NA	6.7×10^{-6}	NA	2.2×10^{-6}	NA	4.8×10^{-9}
Atmospheric population dose (person-rem/year)[c]	3.8×10^{-2}	NA	0.12	NA	9.5×10^{-2}	NA	2.0×10^{-4}
Liquid population dose (person-rem/year)[c]	1.1×10^{-2}	NA	1.4×10^{-2}	NA	3.7×10^{-4}	NA	6.8×10^{-6}
Total population dose (person-rem/year)	4.9×10^{-2}	NA	0.14	NA	9.5×10^{-2}	NA	2.0×10^{-4}
Offsite CO concentration (μg/m^3) - 1-hour average[d]	1.30	NA	3.96	NA	0	NA	0
Offsite CO concentration (μg/m^3) - 8-hour average[d]	0.311	NA	0.947	NA	0	NA	0
Offsite NO_x concentration (μg/m^3) - annual average[d]	1.20×10^{-2}	NA	3.26×10^{-2}	NA	0.113	NA	0
Offsite SO_2 concentration (μg/m^3) - 3-hour average[d]	7.54×10^{-4}	NA	2.32×10^{-3}	NA	0	NA	0
Offsite SO_2 concentration (μg/m^3) - 24-hour average[d]	1.70×10^{-4}	NA	5.21×10^{-4}	NA	0	NA	0
Offsite SO_2 concentration (μg/m^3) - annual average[d]	1.07×10^{-5}	NA	3.27×10^{-5}	NA	0	NA	0
Offsite gaseous fluorides (μg/m^3) 12-hour average[d]	0	NA	0	NA	2.10×10^{-2}	NA	0
Offsite gaseous fluorides (μg/m^3) 24-hour average[d]	0	NA	0	NA	1.13×10^{-2}	NA	0
Offsite gaseous fluorides (μg/m^3) 1-week average[d]	0	NA	0	NA	4.43×10^{-3}	NA	0
Offsite gaseous fluorides (μg/m^3) 1-month average[d]	0	NA	0	NA	125×10^{-3}	NA	0
Offsite HNO_3 concentration (μg/m^3) - 24-hour average[d]	3.21×10^{-3}	NA	8.74×10^{-3}	NA	0.311	NA	0
Offsite HNO_3 concentration (μg/m^3) - annual average[d]	2.01×10^{-4}	NA	5.49×10^{-4}	NA	2.11×10^{-2}	NA	0
Onsite CO concentration (mg/m^3) - 8-hour average[d]	2.08×10^{-3}	NA	6.32×10^{-3}	NA	0	NA	0
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	6.32×10^{-3}	NA	1.72×10^{-2}	NA	7.15×10^{-2}	NA	0
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	2.99×10^{-6}	NA	9.19×10^{-6}	NA	0	NA	0
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	5.65×10^{-5}	NA	1.54×10^{-4}	NA	4.89×10^{-3}	NA	0
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	1.47×10^{-6}	NA	4.57×10^{-6}	NA	0	NA	0
Average number of radiation workers	134	NA	342	NA	82	NA	80
Collective worker dose (person-rem/year)	1.34	NA	13	NA	13	NA	16
Water usage (millions of liters/year)	174	NA	1,089	NA	300	NA	454
Electricity usage (megawatt-hour/year)	13,299	NA	30,233	NA	5,544	NA	6,018
Steam usage (millions of kilograms/year)	79	NA	162	NA	27	NA	6
Fuel usage (thousands of liters/year)	499	NA	1,040	NA	177	NA	19
High-level liquid waste generation (millions of liters/year)	0.12	NA	0.30	NA	0	NA	3.8×10^{-5}
Equivalent DWPF canisters (per year)	2	NA	6	NA	0	NA	0
Saltstone generation (cubic meters/year)	325	NA	825	NA	0	NA	0
TRU waste generation (cubic meters/year)	0	NA	0	NA	0	NA	0
Hazardous/mixed waste generation (cubic meters/year)	0	NA	0	NA	0	NA	0
Low-level waste generation (cubic meters/year)	556	NA	1,456	NA	116	NA	5

TC

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

Table D-22. Data for impact analysis of processing H-Canyon plutonium-239 solutions to metal.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	9.4×10^{-7}	NA	2.8×10^{-6}	NA	4.10×10^{-6}	6.6×10^{-9}	4.4×10^{-9}
Liquid MEI dose (rem/year)	3.7×10^{-6}	NA	3.9×10^{-6}	NA	2.4×10^{-7}	7.3×10^{-10}	4.3×10^{-10}
Total MEI dose (rem/year)	4.7×10^{-6}	NA	6.7×10^{-6}	NA	4.2×10^{-6}	7.3×10^{-9}	4.8×10^{-9}
Atmospheric population dose (person-rem/year)[c]	3.8×10^{-2}	NA	0.12	NA	0.18	2.9×10^{-4}	2.0×10^{-4}
Liquid population dose (person-rem/year)[c]	1.1×10^{-2}	NA	1.4×10^{-2}	NA	3.2×10^{-3}	6.4×10^{-6}	6.8×10^{-6}
Total population dose (person-rem/year)	4.9×10^{-2}	NA	0.14	NA	0.18	3.0×10^{-4}	2.0×10^{-4}
Offsite CO concentration ($\mu g/m^3$) - 1-hour average[d]	1.30	NA	1.30	NA	0	0	0
Offsite CO concentration ($\mu g/m^3$) - 8-hour average[d]	0.311	NA	0.311	NA	0	0	0
Offsite NO_x concentration ($\mu g/m^3$) - annual average[d]	1.2×10^{-2}	NA	9.58×10^{-2}	NA	0.140	0	0
Offsite SO_2 concentration ($\mu g/m^3$) - 3-hour average[d]	7.54×10^{-4}	NA	7.54×10^{-4}	NA	0	0	0
Offsite SO_2 concentration ($\mu g/m^3$) - 24-hour average[d]	1.70×10^{-4}	NA	1.70×10^{-4}	NA	0	0	0
Offsite SO_2 concentration ($\mu g/m^3$) - annual average[d]	1.07×10^{-5}	NA	1.07×10^{-5}	NA	0	0	0
Offsite gaseous fluorides ($\mu g/m^3$) 12-hour average[d]	0	NA	1.55×10^{-2}	NA	2.60×10^{-2}	0	0
Offsite gaseous fluorides ($\mu g/m^3$) 24-hour average[d]	0	NA	8.35×10^{-3}	NA	1.40×10^{-2}	0	0
Offsite gaseous fluorides ($\mu g/m^3$) 1-week average[d]	0	NA	3.28×10^{-3}	NA	5.50×10^{-3}	0	0
Offsite gaseous fluorides ($\mu g/m^3$) 1-month average[d]	0	NA	9.25×10^{-4}	NA	1.55×10^{-3}	0	0
Offsite HNO_3 concentration ($\mu g/m^3$) - 24-hour average[d]	3.21×10^{-3}	NA	0.233	NA	0.385	0	0
Offsite HNO_3 concentration ($\mu g/m^3$) - annual average[d]	2.01×10^{-4}	NA	1.58×10^{-2}	NA	2.62×10^{-2}	0	0
Onsite CO concentration (mg/m^3) - 8-hour average[d]	2.08×10^{-3}	NA	2.08×10^{-3}	NA	0	0	0
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	6.32×10^{-3}	NA	5.93×10^{-2}	NA	8.87×10^{-2}	0	0
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	2.99×10^{-6}	NA	2.99×10^{-6}	NA	0	0	0
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	5.65×10^{-5}	NA	3.67×10^{-3}	NA	6.06×10^{-3}	NA	NA
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	1.47×10^{-6}	NA	1.47×10^{-6}	NA	0	0	0
Average number of radiation workers	134	NA	342	NA	589	157	80
Collective worker dose (person-rem/year)	1.3	NA	13	NA	120	31	16
Water usage (millions of liters/year)	174	NA	1,089	NA	1,216	500	454
Electricity usage (megawatt-hour/year)	13,299	NA	30,233	NA	22,478	6,620	6,018
Steam usage (millions of kilograms/year)	79	NA	162	NA	110	7	6
Fuel usage (thousands of liters/year)	499	NA	1,040	NA	718	21	19
High-level liquid waste generation (millions of liters per year)	0.12	NA	0.30	NA	0.62	0	0
Equivalent DWPF canisters (per year)	2	NA	6	NA	13	0	0
Saltstone generation (cubic meters/year)	325	NA	825	NA	1,689	0	0
TRU waste generation (cubic meters/year)	0	NA	0	NA	64	1	0
Hazardous/mixed waste generation (cubic meters/year)	0	NA	0	NA	0	1	0
Low-level waste generation (cubic meters/year)	556	NA	1,456	NA	5,771	7	5

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995b).

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Table D-23. Data for impact analysis of continuing storage (No Action) of H-Canyon plutonium-239 solutions.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	9.4×10^{-7}	NA	NA	NA	NA	NA	NA
Liquid MEI dose (rem/year)	3.7×10^{-6}	NA	NA	NA	NA	NA	NA
Total MEI dose (rem/year)	4.7×10^{-6}	NA	NA	NA	NA	NA	NA
Atmospheric population dose (person-rem/year)[c]	3.8×10^{-2}	NA	NA	NA	NA	NA	NA
Liquid population dose (person-rem/year)[c]	1.1×10^{-2}	NA	NA	NA	NA	NA	NA
Total population dose (person-rem/year)	4.9×10^{-2}	NA	NA	NA	NA	NA	NA
Offsite CO concentration (μg/m^3) - 1-hour average[d]	1.30	NA	NA	NA	NA	NA	NA
Offsite CO concentration (μg/m^3) - 8-hour average[d]	0.311	NA	NA	NA	NA	NA	NA
Offsite NO_x concentration (μg/m^3) - annual average[d]	1.2×10^{-2}	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration (μg/m^3) - 3-hour average[d]	7.54×10^{-4}	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration (μg/m^3) - 24-hour average[d]	1.70×10^{-4}	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration (μg/m^3) - annual average[d]	1.07×10^{-5}	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 12-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 24-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 1-week average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 1-month average[d]	0	NA	NA	NA	NA	NA	NA
Offsite HNO_3 concentration (μg/m^3) - 24-hour average[d]	3.21×10^{-3}	NA	NA	NA	NA	NA	NA
Offsite HNO_3 concentration (μg/m^3) - annual average[d]	2.01×10^{-4}	NA	NA	NA	NA	NA	NA
Onsite CO concentration (mg/m^3) - 8-hour average[d]	2.08×10^{-3}	NA	NA	NA	NA	NA	NA
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	6.32×10^{-3}	NA	NA	NA	NA	NA	NA
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	2.99×10^{-6}	NA	NA	NA	NA	NA	NA
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	5.65×10^{-5}	NA	NA	NA	NA	NA	NA
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	1.47×10^{-6}	NA	NA	NA	NA	NA	NA
Average number of radiation workers	134	NA	NA	NA	NA	NA	NA
Collective worker dose (person-rem/year)	1.34	NA	NA	NA	NA	NA	NA
Water usage (millions of liters/year)	174	NA	NA	NA	NA	NA	NA
Electricity usage (megawatt-hour/year)	13,299	NA	NA	NA	NA	NA	NA
Steam usage (millions of kilograms/year)	79	NA	NA	NA	NA	NA	NA
Fuel usage (thousands of liters/year)	499	NA	NA	NA	NA	NA	NA
High-level liquid waste generation (millions of liters per year)	0.12	NA	NA	NA	NA	NA	NA
Equivalent DWPF canisters (per year)	2	NA	NA	NA	NA	NA	NA
Saltstone generation (cubic meters/year)	325	NA	NA	NA	NA	NA	NA
TRU waste generation (cubic meters/year)	0	NA	NA	NA	NA	NA	NA
Hazardous/mixed waste generation (cubic meters/year)	0	NA	NA	NA	NA	NA	NA
Low-level waste generation (cubic meters/year)	556	NA	NA	NA	NA	NA	NA

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-24. Data for impact analysis of blending H-Canyon enriched uranium solutions to low enriched uranium.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage[a]
Atmospheric MEI dose (rem/year)	1.4×10^{-6}	NA	6.6×10^{-4}	1.4×10^{-7}	8.5×10^{-7}	NA	-
Liquid MEI dose (rem/year)	5.7×10^{-6}	NA	9.8×10^{-6}	5.7×10^{-7}	2.5×10^{-8}	NA	-
Total MEI dose (rem/year)	7.1×10^{-6}	NA	6.7×10^{-4}	7.1×10^{-7}	8.8×10^{-7}	NA	-
Atmospheric population dose (person-rem/year)[c]	5.8×10^{-2}	NA	27	5.8×10^{-3}	3.8×10^{-2}	NA	-
Liquid population dose (person-rem/year)[c]	1.6×10^{-2}	NA	3.3×10^{-2}	1.6×10^{-3}	1.5×10^{-4}	NA	-
Total population dose (person-rem/year)	7.5×10^{-2}	NA	27	7.5×10^{-3}	3.8×10^{-2}	NA	-
Offsite CO concentration ($\mu g/m^3$) - 1-hour average[d]	3.94	NA	10.6	0.349	0	NA	-
Offsite CO concentration ($\mu g/m^3$) - 8-hour average[d]	0.940	NA	2.54	9.41×10^{-2}	0	NA	-
Offsite NO_x concentration ($\mu g/m^3$) - annual average[d]	5.28×10^{-2}	NA	8.25×10^{-2}	5.28×10^{-3}	2.27×10^{-2}	NA	-
Offsite SO_2 concentration ($\mu g/m^3$) - 3-hour average[d]	2.29×10^{-3}	NA	6.21×10^{-3}	2.25×10^{-4}	0	NA	-
Offsite SO_2 concentration ($\mu g/m^3$) - 24-hour average[d]	5.15×10^{-4}	NA	1.40×10^{-3}	5.06×10^{-5}	0	NA	-
Offsite SO_2 concentration ($\mu g/m^3$) - annual average[d]	3.23×10^{-5}	NA	8.77×10^{-5}	3.18×10^{-6}	0	NA	-
Offsite gaseous fluorides ($\mu g/m^3$) 12-hour average[d]	0	NA	7.48×10^{-6}	0	4.20×10^{-3}	NA	-
Offsite gaseous fluorides ($\mu g/m^3$) 24-hour average[d]	0	NA	2.98×10^{-6}	0	2.26×10^{-3}	NA	-
Offsite gaseous fluorides ($\mu g/m^3$) 1-week average[d]	0	NA	1.58×10^{-6}	0	8.88×10^{-4}	NA	-
Offsite gaseous fluorides ($\mu g/m^3$) 1-month average[d]	0	NA	4.46×10^{-7}	0	2.50×10^{-4}	NA	-
Offsite HNO_3 concentration ($\mu g/m^3$) - 24-hour average[d]	1.42×10^{-2}	NA	2.21×10^{-2}	1.41×10^{-3}	6.22×10^{-2}	NA	-
Offsite HNO_3 concentration ($\mu g/m^3$) - annual average[d]	8.89×10^{-4}	NA	1.39×10^{-3}	8.88×10^{-5}	4.22×10^{-3}	NA	-
Onsite CO concentration (mg/m^3) - 8-hour average[d]	6.28×10^{-3}	NA	1.70×10^{-2}	6.28×10^{-4}	0	NA	-
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	2.79×10^{-2}	NA	4.36×10^{-2}	2.79×10^{-3}	1.43×10^{-2}	NA	-
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	9.08×10^{-6}	NA	2.46×10^{-5}	8.93×10^{-7}	0	NA	-
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	2.50×10^{-4}	NA	3.90×10^{-4}	2.49×10^{-5}	9.77×10^{-4}	NA	-
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	4.52×10^{-6}	NA	1.22×10^{-5}	4.73×10^{-7}	0	NA	-
Average number of radiation workers	232	NA	351	23	33	NA	-
Collective worker dose (person-rem/year)	2.32	NA	14	0.23	5.1	NA	-
Water usage (millions of liters/year)	235	NA	308	24	120	NA	-
Electricity usage (megawatt-hour/year)	18,018	NA	23,595	1,802	2,218	NA	-
Steam usage (millions of kilograms/year)	107	NA	140	11	11	NA	-
Fuel usage (thousands of liters/year)	676	NA	886	68	71	NA	-
High-level liquid waste generation (millions of liters per year)	0.18	NA	2.0	1.8×10^{-2}	0.14	NA	-
Equivalent DWPF canisters (per year)	3	NA	15	0	3	NA	-
Saltstone generation (cubic meters/year)	495	NA	5,570	50	388	NA	-
TRU waste generation (cubic meters/year)	0	NA	0	0	0	NA	-
Hazardous/mixed waste generation (cubic meters/year)	0	NA	0	0	0	NA	-
Low-level waste generation (cubic meters/year)	631	NA	1,029	63	116	NA	-

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

Table D-25. Data for impact analysis of processing H-Canyon enriched uranium solutions to oxide.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	1.4×10^{-6}	NA	6.6×10^{-4}	1.4×10^{-7}	1.0×10^{-11}	NA	2.1×10^{-12}
Liquid MEI dose (rem/year)	5.7×10^{-6}	NA	9.8×10^{-6}	5.7×10^{-7}	0	NA	0
Total MEI dose (rem/year)	7.1×10^{-6}	NA	6.7×10^{-4}	7.1×10^{-7}	1.0×10^{-11}	NA	2.1×10^{-12}
Atmospheric population dose (person-rem/year)[c]	5.8×10^{-2}	NA	27	5.8×10^{-3}	4.2×10^{-7}	NA	8.4×10^{-8}
Liquid population dose (person-rem/year)[c]	1.6×10^{-2}	NA	3.3×10^{-2}	1.6×10^{-3}	0	NA	0
Total population dose (person-rem/year)	7.5×10^{-2}	NA	27	7.5×10^{-3}	4.2×10^{-7}	NA	8.4×10^{-8}
Offsite CO concentration (μg/m^3) - 1-hour average[d]	3.94	NA	10.6	0.349	0	NA	0
Offsite CO concentration (μg/m^3) - 8-hour average[d]	0.940	NA	2.54	9.41×10^{-2}	0	NA	0
Offsite NO_x concentration (μg/m^3) - annual average[d]	5.28×10^{-2}	NA	8.25×10^{-2}	5.28×10^{-3}	2.82×10^{-3}	NA	5.63×10^{-4}
Offsite SO_2 concentration (μg/m^3) - 3-hour average[d]	2.29×10^{-3}	NA	6.21×10^{-3}	2.25×10^{-4}	0	NA	0
Offsite SO_2 concentration (μg/m^3) - 24-hour average[d]	5.15×10^{-4}	NA	1.40×10^{-3}	5.06×10^{-5}	0	NA	0
Offsite SO_2 concentration (μg/m^3) - annual average[d]	3.23×10^{-5}	NA	8.77×10^{-5}	3.18×10^{-6}	0	NA	0
Offsite gaseous fluorides (μg/m^3) 12-hour average[d]	0	NA	7.48×10^{-6}	0	0	NA	0
Offsite gaseous fluorides (μg/m^3) 24-hour average[d]	0	NA	2.98×10^{-6}	0	0	NA	0
Offsite gaseous fluorides (μg/m^3) 1-week average[d]	0	NA	1.58×10^{-6}	0	0	NA	0
Offsite gaseous fluorides (μg/m^3) 1-month average[d]	0	NA	4.46×10^{-7}	0	0	NA	0
Offsite HNO_3 concentration (μg/m^3) - 24-hour average[d]	1.42×10^{-2}	NA	2.21×10^{-2}	1.41×10^{-2}	0	NA	0
Offsite HNO_3 concentration (μg/m^3) - annual average[d]	8.89×10^{-4}	NA	1.39×10^{-3}	8.88×10^{-5}	0	NA	0
Onsite CO concentration (mg/m^3) - 8-hour average[d]	6.28×10^{-3}	NA	1.70×10^{-2}	6.28×10^{-4}	0	NA	0
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	2.79×10^{-2}	NA	4.36×10^{-2}	2.79×10^{-3}	1.49×10^{-3}	NA	2.98×10^{-4}
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	9.08×10^{-6}	NA	2.46×10^{-5}	8.93×10^{-7}	0	NA	0
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	2.50×10^{-4}	NA	3.90×10^{-4}	2.49×10^{-5}	0	NA	0
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	4.52×10^{-6}	NA	1.22×10^{-5}	4.73×10^{-7}	0	NA	0
Average number of radiation workers	232	NA	351	23	80	NA	16
Collective worker dose (person-rem/year)	2.32	NA	14	0.23	51	NA	0.16
Water usage (millions of liters/year)	235	NA	308	24	70	NA	14
Electricity usage (megawatt-hour/year)	18,018	NA	23,595	1,802	5,363	NA	1,073
Steam usage (millions of kilograms/year)	107	NA	140	11	32	NA	6
Fuel usage (thousands of liters/year)	676	NA	886	68	201	NA	40
High-level liquid waste generation (millions of liters per year)	0.18	NA	2.0	1.8×10^{-2}	0	NA	0
Equivalent DWPF canisters (per year)	3	NA	15	0	0	NA	0
Saltstone generation (cubic meters/year)	495	NA	5,570	50	0	NA	0
TRU waste generation (cubic meters/year)	0	NA	0	0	0	NA	0
Hazardous/mixed waste generation (cubic meters/year)	0	NA	0	0	0	NA	0
Low-level waste generation (cubic meters/year)	631	NA	1,029	63	103	NA	21

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

Table D-26. Data for impact analysis of processing and storage for vitrification (DWPF) of H-Canyon enriched uranium solutions.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage[c]	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage[d]
Atmospheric MEI dose (rem/year)	1.4×10^{-6}	NA[d]	1.8×10^{-6}	-	1.6×10^{-4}	NA	-
Liquid MEI dose (rem/year)	5.7×10^{-6}	NA	7.1×10^{-6}	-	2.4×10^{-5}	NA	-
Total MEI dose (rem/year)	7.1×10^{-6}	NA	8.9×10^{-6}	-	1.8×10^{-4}	NA	-
Atmospheric population dose (person-rem/year)[e]	5.8×10^{-2}	NA	7.3×10^{-2}	-	6.9	NA	-
Liquid population dose (person-rem/year)[e]	1.6×10^{-2}	NA	2.1×10^{-2}	-	0.14	NA	-
Total population dose (person-rem/year)	7.5×10^{-2}	NA	9.4×10^{-2}	-	7	NA	-
Offsite CO concentration (μg/m^3) - 1-hour average[f]	3.94	NA	5.85	-	0	NA	-
Offsite CO concentration (μg/m^3) - 8-hour average[f]	0.940	NA	1.40	-	0	NA	-
Offsite NO$_x$ concentration (μg/m^3) - annual average[f]	5.28×10^{-2}	NA	3.32×10^{-2}	-	0	NA	-
Offsite SO$_2$ concentration (μg/m^3) - 3-hour average[f]	2.29×10^{-3}	NA	3.41×10^{-3}	-	0	NA	-
Offsite SO$_2$ concentration (μg/m^3) - 24-hour average[f]	5.15×10^{-4}	NA	7.68×10^{-4}	-	0	NA	-
Offsite SO$_2$ concentration (μg/m^3) - annual average[f]	3.23×10^{-5}	NA	4.82×10^{-5}	-	0	NA	-
Offsite gaseous fluorides (μg/m^3) 12-hour average[f]	0	NA	7.48×10^{-6}	-	0	NA	-
Offsite gaseous fluorides (μg/m^3) 24-hour average[f]	0	NA	2.98×10^{-6}	-	0	NA	-
Offsite gaseous fluorides (μg/m^3) 1-week average[f]	0	NA	1.58×10^{-6}	-	0	NA	-
Offsite gaseous fluorides (μg/m^3) 1-month average[f]	0	NA	4.46×10^{-7}	-	0	NA	-
Offsite HNO$_3$ concentration (μg/m^3) - 24-hour average[f]	1.42×10^{-2}	NA	2.21×10^{-2}	-	0	NA	-
Offsite HNO$_3$ concentration (μg/m^3) - annual average[f]	8.89×10^{-4}	NA	1.39×10^{-3}	-	0	NA	-
Onsite CO concentration (mg/m^3) - 8-hour average[f]	6.28×10^{-3}	NA	9.33×10^{-3}	-	0	NA	-
Onsite NO$_x$ concentration (mg/m^3) - 1-hour average[f]	2.79×10^{-2}	NA	1.76×10^{-2}	-	0	NA	-
Onsite SO$_2$ concentration (mg/m^3) - 8-hour average[f]	9.08×10^{-6}	NA	1.35×10^{-5}	-	0	NA	-
Onsite HNO$_3$ concentration (mg/m^3) - 8-hour average[f]	2.50×10^{-4}	NA	3.90×10^{-4}	-	0	NA	-
Onsite CO$_2$ concentration (mg/m^3) - 8-hour average[f]	4.52×10^{-6}	NA	6.72×10^{-6}	-	0	NA	-
Average number of radiation workers	232	NA	304	-	1,500	NA	-
Collective worker dose (person-rem/year)	2.32	NA	3	-	60	NA	-
Water usage (millions of liters/year)	235	NA	280	-	1,476	NA	-
Electricity usage (megawatt-hour/year)	18,018	NA	21,450	-	5,0371	NA	-
Steam usage (millions of kilograms/year)	107	NA	128	-	167	NA	-
Fuel usage (thousands of liters/year)	676	NA	805	-	0	NA	-
High-level liquid waste generation (millions of liters/year)	0.18	NA	0.23	-	0	NA	-
Equivalent DWPF canisters (per year)	3	NA	3	-	405	NA	-
Saltstone generation (cubic meters/year)	495	NA	619	-	0	NA	-
TRU waste generation (cubic meters/year)	0	NA	0	-	1,136	NA	-
Hazardous/mixed waste generation (cubic meters/year)	0	NA	0	-	3,055	NA	-
Low-level waste generation (cubic meters/year)	631	NA	686	-	2,731	NA	-

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO$_3$ = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO$_x$ = nitrogen oxides, SO$_2$ = sulfur dioxide; TRU = transuranic.

c. Atmospheric releases from the waste tank farms in 1993 contributed less than 1×10^{-8} rem to the maximally exposed individual and less than 50 person-rem to the worker collective dose. At this time, DOE does not expect any discernible increase to these or other environmental factors as a result of this phase in the stabilization alternative.

d. DOE does not expect any discernible incremental impacts as a result of this phase in the stabilization alternative.

e. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

f. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-27. Data for impact analysis of continuing storage (No Action) of H-Canyon enriched uranium solutions.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	1.4×10^{-6}	NA	NA	NA	NA	NA	NA
Liquid MEI dose (rem/year)	5.7×10^{-6}	NA	NA	NA	NA	NA	NA
Total MEI dose (rem/year)	7.1×10^{-6}	NA	NA	NA	NA	NA	NA
Atmospheric population dose (person-rem/year)[c]	5.8×10^{-2}	NA	NA	NA	NA	NA	NA
Liquid population dose (person-rem/year)[c]	1.6×10^{-2}	NA	NA	NA	NA	NA	NA
Total population dose (person-rem/year)	7.5×10^{-2}	NA	NA	NA	NA	NA	NA
Offsite CO concentration (µg/m^3) - 1-hour average[d]	3.94	NA	NA	NA	NA	NA	NA
Offsite CO concentration (µg/m^3) - 8-hour average[d]	0.940	NA	NA	NA	NA	NA	NA
Offsite NO$_x$ concentration (µg/m^3) - annual average[d]	5.28×10^{-2}	NA	NA	NA	NA	NA	NA
Offsite SO$_2$ concentration (µg/m^3) - 3-hour average[d]	2.29×10^{-3}	NA	NA	NA	NA	NA	NA
Offsite SO$_2$ concentration (µg/m^3) - 24-hour average[d]	5.15×10^{-4}	NA	NA	NA	NA	NA	NA
Offsite SO$_2$ concentration (µg/m^3) - annual average[d]	3.23×10^{-5}	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (µg/m^3) 12-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (µg/m^3) 24-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (µg/m^3) 1-week average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (µg/m^3) 1-month average[d]	0	NA	NA	NA	NA	NA	NA
Offsite HNO$_3$ concentration (µg/m^3) - 24-hour average[d]	1.42×10^{-2}	NA	NA	NA	NA	NA	NA
Offsite HNO$_3$ concentration (µg/m^3) - annual average[d]	8.89×10^{-4}	NA	NA	NA	NA	NA	NA
Onsite CO concentration (mg/m^3) - 8-hour average[d]	6.28×10^{-3}	NA	NA	NA	NA	NA	NA
Onsite NO$_x$ concentration (mg/m^3) - 1-hour average[d]	2.79×10^{-2}	NA	NA	NA	NA	NA	NA
Onsite SO$_2$ concentration (mg/m^3) - 8-hour average[d]	9.08×10^{-6}	NA	NA	NA	NA	NA	NA
Onsite HNO$_3$ concentration (mg/m^3) - 8-hour average[d]	2.50×10^{-4}	NA	NA	NA	NA	NA	NA
Onsite CO$_2$ concentration (mg/m^3) - 8-hour average[d]	4.52×10^{-6}	NA	NA	NA	NA	NA	NA
Average number of radiation workers	232	NA	NA	NA	NA	NA	NA
Collective worker dose (person-rem/year)	2.32	NA	NA	NA	NA	NA	NA
Water usage (millions of liters/year)	235	NA	NA	NA	NA	NA	NA
Electricity usage (megawatt-hour/year)	18,018	NA	NA	NA	NA	NA	NA
Steam usage (millions of kilograms/year)	107	NA	NA	NA	NA	NA	NA
Fuel usage (thousands of liters/year)	676	NA	NA	NA	NA	NA	NA
High-level liquid waste generation (millions of liters per year)	0.18	NA	NA	NA	NA	NA	NA
Equivalent DWPF canisters (per year)	3	NA	NA	NA	NA	NA	NA
Saltstone generation (cubic meters/year)	495	NA	NA	NA	NA	NA	NA
TRU waste generation (cubic meters/year)	0	NA	NA	NA	NA	NA	NA
Hazardous/mixed waste generation (cubic meters/year)	0	NA	NA	NA	NA	NA	NA
Low-level waste generation (cubic meters/year)	631	NA	NA	NA	NA	NA	NA

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO$_3$ = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO$_x$ = nitrogen oxides, SO$_2$ = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-28. Data for impact analysis of improving storage of plutonium and uranium stored in vaults.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	7.6×10^{-9}	7.4×10^{-7}	NA	7.6×10^{-9}	NA	6.6×10^{-9}	4.4×10^{-9}
Liquid MEI dose (rem/year)	7.4×10^{-10}	2.0×10^{-8}	NA	1.2×10^{-5}	NA	6.4×10^{-10}	6.8×10^{-8}
Total MEI dose (rem/year)	8.3×10^{-9}	7.6×10^{-7}	NA	8.3×10^{-9}	NA	7.3×10^{-9}	4.8×10^{-9}
Atmospheric population dose (person-rem/year)[c]	3.4×10^{-4}	3.3×10^{-2}	NA	3.4×10^{-4}	NA	2.9×10^{-4}	2.0×10^{-4}
Liquid population dose (person-rem/year)[c]	1.2×10^{-5}	3.1×10^{-4}	NA	1.2×10^{-5}	NA	6.4×10^{-6}	6.8×10^{-6}
Total population dose (person-rem/year)	3.5×10^{-4}	3.3×10^{-2}	NA	3.5×10^{-4}	NA	3.0×10^{-4}	2.0×10^{-4}
Offsite CO concentration ($\mu g/m^3$) - 1-hour average[d]	0	0	NA	0	NA	0	0
Offsite CO concentration ($\mu g/m^3$) - 8-hour average[d]	0	0	NA	0	NA	0	0
Offsite NO_x concentration ($\mu g/m^3$) - annual average[d]	0	2.04×10^{-2}	NA	0	NA	0	0
Offsite SO_2 concentration ($\mu g/m^3$) - 3-hour average[d]	0	0	NA	0	NA	0	0
Offsite SO_2 concentration ($\mu g/m^3$) - 24-hour average[d]	0	0	NA	0	NA	0	0
Offsite SO_2 concentration ($\mu g/m^3$) - annual average[d]	0	0	NA	0	NA	0	0
Offsite gaseous fluorides ($\mu g/m^3$) 12-hour average[d]	0	3.78×10^{-3}	NA	0	NA	0	0
Offsite gaseous fluorides ($\mu g/m^3$) 24-hour average[d]	0	2.03×10^{-3}	NA	0	NA	0	0
Offsite gaseous fluorides ($\mu g/m^3$) 1-week average[d]	0	7.99×10^{-4}	NA	0	NA	0	0
Offsite gaseous fluorides ($\mu g/m^3$) 1-month average[d]	0	2.25×10^{-4}	NA	0	NA	0	0
Offsite HNO_3 concentration ($\mu g/m^3$) - 24-hour average[d]	0	5.60×10^{-2}	NA	0	NA	0	0
Offsite HNO_3 concentration ($\mu g/m^3$) - annual average[d]	0	3.80×10^{-3}	NA	0	NA	0	0
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	0	NA	0	NA	0	0
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	0	1.29×10^{-2}	NA	0	NA	0	0
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	0	NA	0	NA	0	0
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	0	8.79×10^{-4}	NA	0	NA	0	0
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	0	NA	0	NA	0	0
Average number of radiation workers	81	70	NA	81	NA	157	80
Collective worker dose (person-rem/year)	7.7	39	NA	7.7	NA	31	16
Water usage (millions of liters/year)	714	55	NA	714	NA	500	454
Electricity usage (megawatt-hour/year)	9,460	1,008	NA	9,460	NA	6,620	6,018
Steam usage (millions of kilograms/year)	9	5	NA	9	NA	7	6
Fuel usage (thousands of liters/year)	30	32	NA	30	NA	21	19
High-level liquid waste generation (millions of liters/year)	0	0	NA	0	NA	0	0
Equivalent DWPF canisters (per year)	0	0	NA	0	NA	0	0
Saltstone generation (cubic meters/year)	0	0	NA	0	NA	0	0
TRU waste generation (cubic meters/year)	2	26	NA	2	NA	1	0
Hazardous/mixed waste generation (cubic meters/year)	1	0	NA	1	NA	1	0
Low-level waste generation (cubic meters/year)	1,297	1,889	NA	1,297	NA	7	5

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-29. Data for impact analysis of processing plutonium and uranium stored in vaults to metal.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	7.6×10^{-9}	7.4×10^{-7}	8.3×10^{-4}	7.6×10^{-9}	1.1×10^{-5}	6.6×10^{-9}	4.4×10^{-9}
Liquid MEI dose (rem/year)	7.4×10^{-10}	2.0×10^{-8}	1.2×10^{-5}	7.4×10^{-10}	3.1×10^{-7}	7.3×10^{-10}	4.3×10^{-10}
Total MEI dose (rem/year)	8.3×10^{-9}	7.6×10^{-7}	8.4×10^{-4}	8.3×10^{-9}	1.1×10^{-5}	7.3×10^{-9}	4.8×10^{-9}
Atmospheric population dose (person-rem/year)[c]	3.4×10^{-4}	3.3×10^{-2}	34	3.4×10^{-4}	0.48	2.9×10^{-4}	2.0×10^{-4}
Liquid population dose (person-rem/year)[c]	1.2×10^{-5}	3.1×10^{-4}	4.2×10^{-2}	1.2×10^{-5}	1.8×10^{-3}	6.4×10^{-6}	6.8×10^{-6}
Total population dose (person-rem/year)	3.5×10^{-4}	3.3×10^{-2}	34	3.5×10^{-4}	0.48	3.0×10^{-4}	2.0×10^{-4}
Offsite CO concentration ($\mu g/m^3$) - 1-hour average[d]	0	0	13.3	0	0	0	0
Offsite CO concentration ($\mu g/m^3$) - 8-hour average[d]	0	0	3.18	0	0	0	0
Offsite NO_x concentration ($\mu g/m^3$) - annual average[d]	0	2.04×10^{-2}	0.103	0	0.283	0	0
Offsite SO_2 concentration ($\mu g/m^3$) - 3-hour average[d]	0	0	7.75×10^{-3}	0	0	0	0
Offsite SO_2 concentration ($\mu g/m^3$) - 24-hour average[d]	0	0	1.74×10^{-3}	0	0	0	0
Offsite SO_2 concentration ($\mu g/m^3$) - annual average[d]	0	0	1.10×10^{-4}	0	0	0	0
Offsite gaseous fluorides ($\mu g/m^3$) 12-hour average[d]	0	3.78×10^{-3}	1.50×10^{-5}	0	5.24×10^{-2}	0	0
Offsite gaseous fluorides ($\mu g/m^3$) 24-hour average[d]	0	2.03×10^{-3}	5.95×10^{-6}	0	2.82×10^{-2}	0	0
Offsite gaseous fluorides ($\mu g/m^3$) 1-week average[d]	0	7.99×10^{-4}	3.16×10^{-6}	0	1.11×10^{-2}	0	0
Offsite gaseous fluorides ($\mu g/m^3$) 1-month average[d]	0	2.25×10^{-4}	8.92×10^{-7}	0	3.12×10^{-3}	0	0
Offsite HNO_3 concentration ($\mu g/m^3$) - 24-hour average[d]	0	5.60×10^{-2}	2.76×10^{-2}	0	0.777	0	0
Offsite HNO_3 concentration ($\mu g/m^3$) - annual average[d]	0	3.80×10^{-3}	1.74×10^{-3}	0	5.27×10^{-2}	0	0
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	0	2.12×10^{-2}	0	0	0	0
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	0	1.29×10^{-2}	5.45×10^{-2}	0	0.179	0	0
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	0	3.08×10^{-5}	0	0	0	0
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	0	8.79×10^{-4}	4.88×10^{-4}	0	1.22×10^{-2}	0	0
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	0	1.52×10^{-5}	0	0	0	0
Average number of radiation workers	81	70	508	81	708	157	80
Collective worker dose (person-rem/year)	7.7	39	49	7.7	160	31	16
Water usage (millions of liters/year)	714	55	347	714	1,499	500	454
Electricity usage (megawatt-hour/year)	9,460	1,008	29,887	9,460	27,720	6,620	6,018
Steam usage (millions of kilograms/year)	9	5	145	9	136	7	6
Fuel usage (thousands of liters/year)	30	32	907	30	886	21	19
High-level liquid waste generation (millions of liters per year)	0	0	2.0	0	1.9	0	0
Equivalent DWPF canisters (per year)	0	0	15	0	38	0	0
Saltstone generation (cubic meters/year)	0	0	5,570	0	5,077	0	0
TRU waste generation (cubic meters/year)	2	26	98	2	64	1	0
Hazardous/mixed waste generation (cubic meters/year)	1	0	119	1	0	1	0
Low-level waste generation (cubic meters/year)	1,297	1,889	1,799	1,297	6,030	7	5

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

Table D-30. Data for impact analysis of processing plutonium and uranium stored in vaults to oxide.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	7.6×10^{-9}	7.4×10^{-7}	8.3×10^{-4}	7.6×10^{-9}	NA	6.6×10^{-9}	4.4×10^{-9}
Liquid MEI dose (rem/year)	7.4×10^{-10}	2.0×10^{-8}	1.2×10^{-5}	7.4×10^{-10}	NA	7.3×10^{-10}	4.3×10^{-10}
Total MEI dose (rem/year)	8.3×10^{-9}	7.6×10^{-7}	8.4×10^{-4}	8.3×10^{-9}	NA	7.3×10^{-9}	4.8×10^{-9}
Atmospheric population dose (person-rem/year)[c]	3.4×10^{-4}	3.3×10^{-2}	34	3.4×10^{-4}	NA	2.9×10^{-4}	2.0×10^{-4}
Liquid population dose (person-rem/year)[c]	1.2×10^{-5}	3.1×10^{-4}	4.2×10^{-2}	1.2×10^{-5}	NA	6.4×10^{-6}	6.8×10^{-6}
Total population dose (person-rem/year)	3.5×10^{-4}	3.3×10^{-2}	34	3.5×10^{-4}	NA	3.0×10^{-4}	2.0×10^{-4}
Offsite CO concentration (μg/m^3) - 1-hour average[d]	0	0	13.3	0	NA	0	0
Offsite CO concentration (μg/m^3) - 8-hour average[d]	0	0	3.18	0	NA	0	0
Offsite NO_x concentration (μg/m^3) - annual average[d]	0	2.04×10^{-2}	0.103	0	NA	0	0
Offsite SO_2 concentration (μg/m^3) - 3-hour average[d]	0	0	7.75×10^{-3}	0	NA	0	0
Offsite SO_2 concentration (μg/m^3) - 24-hour average[d]	0	0	1.74×10^{-3}	0	NA	0	0
Offsite SO_2 concentration (μg/m^3) - annual average[d]	0	0	1.10×10^{-4}	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 12-hour average[d]	0	3.78×10^{-3}	1.50×10^{-5}	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 24-hour average[d]	0	2.03×10^{-3}	5.95×10^{-6}	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 1-week average[d]	0	7.99×10^{-4}	3.16×10^{-6}	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 1-month average[d]	0	2.25×10^{-4}	8.92×10^{-7}	0	NA	0	0
Offsite HNO_3 concentration (μg/m^3) - 24-hour average[d]	0	5.60×10^{-2}	2.76×10^{-2}	0	NA	0	0
Offsite HNO_3 concentration (μg/m^3) - annual average[d]	0	3.80×10^{-3}	1.74×10^{-3}	0	NA	0	0
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	0	2.12×10^{-2}	0	NA	0	0
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	0	1.29×10^{-2}	5.45×10^{-2}	0	NA	0	0
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	0	3.08×10^{-5}	0	NA	0	0
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	0	8.79×10^{-4}	4.88×10^{-4}	0	NA	0	0
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	0	1.52×10^{-5}	0	NA	0	0
Average number of radiation workers	81	70	508	81	NA	157	80
Collective worker dose (person-rem/year)	7.7	39	49	7.7	NA	31	16
Water usage (millions of liters/year)	714	55	347	714	NA	500	454
Electricity usage (megawatt-hour/year)	9,460	1,008	28,890	9,460	NA	6,620	6,018
Steam usage (millions of kilograms/year)	9	5	145	9	NA	7	6
Fuel usage (thousands of liters/year)	30	32	907	30	NA	21	19
High-level liquid waste generation (millions of liters per year)	0	0	2.0	0	NA	0	0
Equivalent DWPF canisters (per year)	0	0	15	0	NA	0	0
Saltstone generation (cubic meters/year)	0	0	5,570	0	NA	0	0
TRU waste generation (cubic meters/year)	2	26	98	2	NA	1	0
Hazardous/mixed waste generation (cubic meters/year)	1	0	119	1	NA	1	0
Low-level waste generation (cubic meters/year)	1,297	1,889	1,799	1,297	NA	7	5

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

Table D-31. Data for impact analysis of vitrification (F-Canyon) of plutonium and uranium stored in vaults.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	7.6×10^{-9}	7.4×10^{-7}	8.3×10^{-4}	NA	2.1×10^{-6}	NA	4.4×10^{-9}
Liquid MEI dose (rem/year)	7.4×10^{-10}	2.0×10^{-8}	1.2×10^{-5}	NA	6.3×10^{-8}	NA	4.3×10^{-10}
Total MEI dose (rem/year)	8.3×10^{-9}	7.6×10^{-7}	8.4×10^{-4}	NA	2.2×10^{-6}	NA	4.8×10^{-9}
Atmospheric population dose (person-rem/year)[c]	3.4×10^{-4}	3.3×10^{-2}	34	NA	9.5×10^{-2}	NA	2.0×10^{-4}
Liquid population dose (person-rem/year)[c]	1.2×10^{-5}	3.1×10^{-4}	4.2×10^{-2}	NA	3.7×10^{-4}	NA	6.8×10^{-6}
Total population dose (person-rem/year)	3.5×10^{-4}	3.3×10^{-2}	34	NA	9.5×10^{-2}	NA	2.0×10^{-4}
Offsite CO concentration ($\mu g/m^3$) - 1-hour average[d]	0	0	13.3	NA	0	NA	0
Offsite CO concentration ($\mu g/m^3$) - 8-hour average[d]	0	0	3.18	NA	0	NA	0
Offsite NO_x concentration ($\mu g/m^3$) - annual average[d]	0	2.04×10^{-2}	0.103	NA	0.113	NA	0
Offsite SO_2 concentration ($\mu g/m^3$) - 3-hour average[d]	0	0	7.75×10^{-3}	NA	0	NA	0
Offsite SO_2 concentration ($\mu g/m^3$) - 24-hour average[d]	0	0	1.74×10^{-3}	NA	0	NA	0
Offsite SO_2 concentration ($\mu g/m^3$) - annual average[d]	0	0	1.10×10^{-4}	NA	0	NA	0
Offsite gaseous fluorides ($\mu g/m^3$) 12-hour average[d]	0	3.78×10^{-3}	1.50×10^{-5}	NA	2.10×10^{-2}	NA	0
Offsite gaseous fluorides ($\mu g/m^3$) 24-hour average[d]	0	2.03×10^{-3}	5.95×10^{-6}	NA	1.13×10^{-2}	NA	0
Offsite gaseous fluorides ($\mu g/m^3$) 1-week average[d]	0	7.99×10^{-4}	3.16×10^{-6}	NA	4.43×10^{-3}	NA	0
Offsite gaseous fluorides ($\mu g/m^3$) 1-month average[d]	0	2.25×10^{-4}	8.92×10^{-7}	NA	1.25×10^{-3}	NA	0
Offsite HNO_3 concentration ($\mu g/m^3$) - 24-hour average[d]	0	5.60×10^{-2}	2.76×10^{-2}	NA	0.311	NA	0
Offsite HNO_3 concentration ($\mu g/m^3$) - annual average[d]	0	3.80×10^{-2}	1.74×10^{-3}	NA	2.11×10^{-2}	NA	0
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	0	2.12×10^{-2}	NA	0	NA	0
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	0	1.29×10^{-2}	5.45×10^{-2}	NA	7.15×10^{-2}	NA	0
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	0	3.08×10^{-5}	NA	0	NA	0
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	0	8.79×10^{-4}	4.88×10^{-4}	NA	4.89×10^{-3}	NA	0
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	0	1.52×10^{-5}	NA	0	NA	0
Average number of radiation workers	81	70	508	NA	82	NA	80
Collective worker dose (person-rem/year)	7.7	39	49	NA	13	NA	16
Water usage (millions of liters/year)	714	55	347	NA	300	NA	454
Electricity usage (megawatt-hour/year)	9,460	1,008	29,887	NA	5,544	NA	6,018
Steam usage (millions of kilograms/year)	9	5	145	NA	27	NA	6
Fuel usage (thousands of liters/year)	30	32	907	NA	177	NA	19
High-level liquid waste generation (millions of liters per year)	0	0	2.0	NA	0	NA	0
Equivalent DWPF canisters (per year)	0	0	15	NA	0	NA	0
Saltstone generation (cubic meters/year)	0	0	5,570	NA	0	NA	0
TRU waste generation (cubic meters/year)	2	26	98	NA	0	NA	0
Hazardous/mixed waste generation (cubic meters/year)	1	0	119	NA	0	NA	0
Low-level waste generation (cubic meters/year)	1,297	1,889	1,799	NA	116	NA	5

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-32. Data for impact analysis of processing and storage for vitrification (DWPF) of plutonium and uranium stored in vaults.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage[c]	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage[d]
Atmospheric MEI dose (rem/year)	7.6×10^{-9}	NA	8.3×10^{-4}	-	1.6×10^{-4}	NA	-
Liquid MEI dose (rem/year)	7.4×10^{-10}	NA	1.2×10^{-5}	-	2.4×10^{-5}	NA	-
Total MEI dose (rem/year)	8.3×10^{-9}	NA	8.4×10^{-4}	-	1.8×10^{-4}	NA	-
Atmospheric population dose (person-rem/year)[e]	3.4×10^{-4}	NA	34	-	6.9	NA	-
Liquid population dose (person-rem/year)[e]	1.2×10^{-5}	NA	4.2×10^{-2}	-	0.14	NA	-
Total population dose (person-rem/year)	3.5×10^{-4}	NA	34	-	7.0	NA	-
Offsite CO concentration ($\mu g/m^3$) - 1-hour average[f]	0	NA	13.3	-	0	NA	-
Offsite CO concentration ($\mu g/m^3$) - 8-hour average[f]	0	NA	3.18	-	0	NA	-
Offsite NO_x concentration ($\mu g/m^3$) - annual average[f]	0	NA	0.103	-	0	NA	-
Offsite SO_2 concentration ($\mu g/m^3$) - 3-hour average[f]	0	NA	7.75×10^{-3}	-	0	NA	-
Offsite SO_2 concentration ($\mu g/m^3$) - 24-hour average[f]	0	NA	1.74×10^{-3}	-	0	NA	-
Offsite SO_2 concentration ($\mu g/m^3$) - annual average[f]	0	NA	1.10×10^{-4}	-	0	NA	-
Offsite gaseous fluorides ($\mu g/m^3$) 12-hour average[f]	0	NA	1.50×10^{-5}	-	0	NA	-
Offsite gaseous fluorides ($\mu g/m^3$) 24-hour average[f]	0	NA	5.95×10^{-6}	-	0	NA	-
Offsite gaseous fluorides ($\mu g/m^3$) 1-week average[f]	0	NA	3.16×10^{-6}	-	0	NA	-
Offsite gaseous fluorides ($\mu g/m^3$) 1-month average[f]	0	NA	8.92×10^{-7}	-	0	NA	-
Offsite HNO_3 concentration ($\mu g/m^3$) - 24-hour average[f]	0	NA	2.76×10^{-2}	-	0	NA	-
Offsite HNO_3 concentration ($\mu g/m^3$) - annual average[f]	0	NA	1.74×10^{-3}	-	0	NA	-
Onsite CO concentration (mg/m^3) - 8-hour average[f]	0	NA	2.12×10^{-2}	-	0	NA	-
Onsite NO_x concentration (mg/m^3) - 1-hour average[f]	0	NA	5.45×10^{-2}	-	0	NA	-
Onsite SO_2 concentration (mg/m^3) - 8-hour average[f]	0	NA	3.08×10^{-5}	-	0	NA	-
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[f]	0	NA	4.88×10^{-4}	-	0	NA	-
Onsite CO_2 concentration (mg/m^3) - 8-hour average[f]	0	NA	1.52×10^{-5}	-	0	NA	-
Average number of radiation workers	81	NA	508	-	1,500	NA	-
Collective worker dose (person-rem/year)	7.7	NA	49	-	60	NA	-
Water usage (millions of liters/year)	714	NA	347	-	1,476	NA	-
Electricity usage (megawatt-hour/year)	9,460	NA	29,887	-	50,371	NA	-
Steam usage (millions of kilograms/year)	9	NA	145	-	167	NA	-
Fuel usage (thousands of liters/year)	30	NA	907	-	0	NA	-
High-level liquid waste generation (millions of liters per year)	0	NA	2.0	-	0	NA	-
Equivalent DWPF canisters (per year)	0	NA	15	-	405	NA	-
Saltstone generation (cubic meters/year)	0	NA	5,570	-	0	NA	-
TRU waste generation (cubic meters/year)	2	NA	98	-	1,136	NA	-
Hazardous/mixed waste generation (cubic meters/year)	1	NA	119	-	3,055	NA	-
Low-level waste generation (cubic meters/year)	1,297	NA	1,799	-	2,731	NA	-

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Atmospheric releases from the waste tank farms in 1993 contributed less than 1×10^{-8} rem to the maximally exposed individual and less than 50 person-rem to the worker collective dose. At this time, DOE does not expect any discernible increase to these or other environmental factors as a result of this phase in the stabilization alternative.

d. DOE does not expect any discernible incremental impacts as a result of this phase in the stabilization alternative.

e. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

f. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

Table D-33. Data for impact analysis of continuing storage (No Action) of plutonium and uranium stored in vaults.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	7.6×10^{-9}	NA	NA	NA	NA	NA	NA
Liquid MEI dose (rem/year)	7.4×10^{-10}	NA	NA	NA	NA	NA	NA
Total MEI dose (rem/year)	8.3×10^{-9}	NA	NA	NA	NA	NA	NA
Atmospheric population dose (person-rem/year)[c]	3.4×10^{-4}	NA	NA	NA	NA	NA	NA
Liquid population dose (person-rem/year)[c]	1.2×10^{-5}	NA	NA	NA	NA	NA	NA
Total population dose (person-rem/year)	3.5×10^{-4}	NA	NA	NA	NA	NA	NA
Offsite CO concentration (μg/m^3) - 1-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite CO concentration (μg/m^3) - 8-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite NO_x concentration (μg/m^3) - annual average[d]	0	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration (μg/m^3) - 3-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration (μg/m^3) - 24-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration (μg/m^3) - annual average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 12-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 24-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 1-week average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 1-month average[d]	0	NA	NA	NA	NA	NA	NA
Offsite HNO_3 concentration (μg/m^3) - 24-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite HNO_3 concentration (μg/m^3) - annual average[d]	0	NA	NA	NA	NA	NA	NA
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	NA	NA	NA	NA
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	0	NA	NA	NA	NA	NA	NA
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	NA	NA	NA	NA
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	NA	NA	NA	NA
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	NA	NA	NA	NA
Average number of radiation workers	81	NA	NA	NA	NA	NA	NA
Collective worker dose (person-rem/year)	7.7	NA	NA	NA	NA	NA	NA
Water usage (millions of liters/year)	714	NA	NA	NA	NA	NA	NA
Electricity usage (megawatt-hour/year)	9,460	NA	NA	NA	NA	NA	NA
Steam usage (millions of kilograms/year)	9	NA	NA	NA	NA	NA	NA
Fuel usage (thousands of liters/year)	30	NA	NA	NA	NA	NA	NA
High-level liquid waste generation (millions of liters per year)	0	NA	NA	NA	NA	NA	NA
Equivalent DWPF canisters (per year)	0	NA	NA	NA	NA	NA	NA
Saltstone generation (cubic meters/year)	0	NA	NA	NA	NA	NA	NA
TRU waste generation (cubic meters/year)	2	NA	NA	NA	NA	NA	NA
Hazardous/mixed waste generation (cubic meters/year)	1	NA	NA	NA	NA	NA	NA
Low-level waste generation (cubic meters/year)	1,297	NA	NA	NA	NA	NA	NA

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-34. Data for impact analysis of improving storage of plutonium and uranium stored in vaults (plutonium-238 scrap material).[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	4.0×10^{-7}	4.5×10^{-7}	NA	4.0×10^{-7}	NA	NA	NA
Liquid MEI dose (rem/year)	1.6×10^{-6}	1.8×10^{-6}	NA	1.6×10^{-6}	NA	NA	NA
Total MEI dose (rem/year)	2.0×10^{-6}	2.2×10^{-6}	NA	2.0×10^{-6}	NA	NA	NA
Atmospheric population dose (person-rem/year)[c]	1.6×10^{-2}	1.8×10^{-2}	NA	1.6×10^{-2}	NA	NA	NA
Liquid population dose (person-rem/year)[c]	4.6×10^{-3}	5.1×10^{-3}	NA	4.6×10^{-3}	NA	NA	NA
Total population dose (person-rem/year)	2.1×10^{-2}	2.3×10^{-2}	NA	2.1×10^{-2}	NA	NA	NA
Offsite CO concentration ($\mu g/m^3$) - 1-hour average[d]	1.22	1.46	NA	1.22	NA	NA	NA
Offsite CO concentration ($\mu g/m^3$) - 8-hour average[d]	0.292	0.350	NA	0.292	NA	NA	NA
Offsite NO_x concentration ($\mu g/m^3$) - annual average[d]	9.49×10^{-3}	1.1×10^{-2}	NA	9.49×10^{-3}	NA	NA	NA
Offsite SO_2 concentration ($\mu g/m^3$) - 3-hour average[d]	7.15×10^{-4}	8.47×10^{-4}	NA	7.15×10^{-4}	NA	NA	NA
Offsite SO_2 concentration ($\mu g/m^3$) - 24-hour average[d]	1.61×10^{-4}	1.91×10^{-4}	NA	1.61×10^{-4}	NA	NA	NA
Offsite SO_2 concentration ($\mu g/m^3$) - annual average[d]	1.01×10^{-5}	1.20×10^{-5}	NA	1.01×10^{-5}	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 12-hour average[d]	0	0	NA	0	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 24-hour average[d]	0	0	NA	0	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 1-week average[d]	0	0	NA	0	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 1-month average[d]	0	0	NA	0	NA	NA	NA
Offsite HNO_3 concentration ($\mu g/m^3$) - 24-hour average[d]	2.54×10^{-3}	3.04×10^{-3}	NA	2.54×10^{-3}	NA	NA	NA
Offsite HNO_3 concentration ($\mu g/m^3$) - annual average[d]	1.60×10^{-4}	1.91×10^{-4}	NA	1.60×10^{-4}	NA	NA	NA
Onsite CO concentration (mg/m^3) - 8-hour average[d]	1.95×10^{-3}	2.33×10^{-3}	NA	1.95×10^{-3}	NA	NA	NA
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	5.01×10^{-3}	6.00×10^{-3}	NA	5.01×10^{-3}	NA	NA	NA
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	2.84×10^{-6}	3.36×10^{-7}	NA	2.84×10^{-6}	NA	NA	NA
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	4.48×10^{-5}	5.36×10^{-5}	NA	4.48×10^{-5}	NA	NA	NA
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	1.42×10^{-6}	1.68×10^{-6}	NA	1.42×10^{-6}	NA	NA	NA
Average number of radiation workers	78	89	NA	78	NA	NA	NA
Collective worker dose (person-rem/year)	5.76	7.12	NA	5.76	NA	NA	NA
Water usage (millions of liters/year)	32	35	NA	32	NA	NA	NA
Electricity usage (megawatt-hour/year)	5,262	5,720	NA	5,262	NA	NA	NA
Steam usage (millions of kilograms/year)	4	4	NA	4	NA	NA	NA
Fuel usage (thousands of liters/year)	17	19	NA	17	NA	NA	NA
High-level liquid waste generation (millions of liters per year)	0	0	NA	0	NA	NA	NA
Equivalent DWPF canisters (per year)	0	0	NA	0	NA	NA	NA
Saltstone generation (cubic meters/year)	0	0	NA	0	NA	NA	NA
TRU waste generation (cubic meters/year)	79	82	NA	79	NA	NA	NA
Hazardous/mixed waste generation (cubic meters/year)	96	99	NA	96	NA	NA	NA
Low-level waste generation (cubic meters/year)	617	642	NA	617	NA	NA	NA

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-35. Data for impact analysis of processing plutonium and uranium stored in vaults (plutonium-238 scrap material) to oxide.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	4.0×10^{-7}	NA	1.7×10^{-4}	4.0×10^{-7}	NA	NA	NA
Liquid MEI dose (rem/year)	1.6×10^{-6}	NA	6.5×10^{-6}	1.6×10^{-6}	NA	NA	NA
Total MEI dose (rem/year)	2.0×10^{-6}	NA	1.7×10^{-4}	2.0×10^{-6}	NA	NA	NA
Atmospheric population dose (person-rem/year)[c]	1.6×10^{-2}	NA	6.8	1.6×10^{-2}	NA	NA	NA
Liquid population dose (person-rem/year)[c]	4.6×10^{-3}	NA	2.0×10^{-2}	4.6×10^{-3}	NA	NA	NA
Total population dose (person-rem/year)	2.1×10^{-2}	NA	6.8	2.1×10^{-2}	NA	NA	NA
Offsite CO concentration (μg/m^3) - 1-hour average[d]	1.22	NA	4.40	1.22	NA	NA	NA
Offsite CO concentration (μg/m^3) - 8-hour average[d]	0.292	NA	1.06	0.292	NA	NA	NA
Offsite NO_x concentration (μg/m^3) - annual average[d]	9.49×10^{-3}	NA	4.0×10^{-2}	9.49×10^{-3}	NA	NA	NA
Offsite SO_2 concentration (μg/m^3) - 3-hour average[d]	7.15×10^{-4}	NA	2.59×10^{-3}	7.15×10^{-4}	NA	NA	NA
Offsite SO_2 concentration (μg/m^3) - 24-hour average[d]	1.61×10^{-4}	NA	5.84×10^{-4}	1.61×10^{-4}	NA	NA	NA
Offsite SO_2 concentration (μg/m^3) - annual average[d]	1.01×10^{-5}	NA	3.66×10^{-5}	1.01×10^{-5}	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 12-hour average[d]	0	NA	0	0	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 24-hour average[d]	0	NA	0	0	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 1-week average[d]	0	NA	0	0	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 1-month average[d]	0	NA	0	0	NA	NA	NA
Offsite HNO_3 concentration (μg/m^3) - 24-hour average[d]	2.54×10^{-3}	NA	1.70×10^{-2}	2.54×10^{-3}	NA	NA	NA
Offsite HNO_3 concentration (μg/m^3) - annual average[d]	1.60×10^{-4}	NA	6.74×10^{-4}	1.60×10^{-4}	NA	NA	NA
Onsite CO concentration (mg/m^3) - 8-hour average[d]	1.95×10^{-3}	NA	7.08×10^{-3}	1.95×10^{-3}	NA	NA	NA
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	5.01×10^{-3}	NA	2.11×10^{-2}	5.01×10^{-3}	NA	NA	NA
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	2.84×10^{-6}	NA	1.03×10^{-5}	2.84×10^{-6}	NA	NA	NA
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	4.48×10^{-5}	NA	1.89×10^{-4}	4.48×10^{-5}	NA	NA	NA
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	1.42×10^{-6}	NA	5.06×10^{-6}	1.42×10^{-6}	NA	NA	NA
Average number of radiation workers	78	NA	309	78	NA	NA	NA
Collective worker dose (person-rem/year)	5.76	NA	36	5.76	NA	NA	NA
Water usage (millions of liters/year)	32	NA	223	32	NA	NA	NA
Electricity usage (megawatt-hour/year)	5,262	NA	20,449	5,262	NA	NA	NA
Steam usage (millions of kilograms/year)	4	NA	89	4	NA	NA	NA
Fuel usage (thousands of liters/year)	17	NA	552	17	NA	NA	NA
High-level liquid waste generation (millions of liters per year)	0	NA	0.13	0	NA	NA	NA
Equivalent DWPF canisters (per year)	0	NA	2	0	NA	NA	NA
Saltstone generation (cubic meters/year)	0	NA	356	0	NA	NA	NA
TRU waste generation (cubic meters/year)	79	NA	98	79	NA	NA	NA
Hazardous/mixed waste generation (cubic meters/year)	96	NA	119	96	NA	NA	NA
Low-level waste generation (cubic meters/year)	617	NA	1,340	617	NA	NA	NA

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-36. Data for impact analysis of processing and storage for vitrification (DWPF) of plutonium and uranium stored in vaults (plutonium-238 scrap material).[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage[c]	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage[d]
Atmospheric MEI dose (rem/year)	4.0×10^{-7}	NA	1.4×10^{-6}	-	1.6×10^{-4}	NA	-
Liquid MEI dose (rem/year)	1.6×10^{-6}	NA	5.6×10^{-6}	-	2.3×10^{-5}	NA	-
Total MEI dose (rem/year)	2.0×10^{-6}	NA	7.1×10^{-6}	-	1.8×10^{-4}	NA	-
Atmospheric population dose (person-rem/year)[e]	1.6×10^{-2}	NA	5.8×10^{-2}	-	6.3	NA	-
Liquid population dose (person-rem/year)[e]	4.6×10^{-3}	NA	1.6×10^{-2}	-	6.6×10^{-2}	NA	-
Total population dose (person-rem/year)	2.1×10^{-2}	NA	7.5×10^{-2}	-	6.4	NA	-
Offsite CO concentration (μg/m^3) - 1-hour average[f]	1.22	NA	3.01	-	0	NA	-
Offsite CO concentration (μg/m^3) - 8-hour average[f]	0.292	NA	0.718	-	0	NA	-
Offsite NO_x concentration (μg/m^3) - annual average[f]	9.49×10^{-3}	NA	2.89×10^{-2}	-	0	NA	-
Offsite SO_2 concentration (μg/m^3) - 3-hour average[f]	7.15×10^{-4}	NA	1.75×10^{-3}	-	0	NA	-
Offsite SO_2 concentration (μg/m^3) - 24-hour average[f]	1.61×10^{-4}	NA	3.93×10^{-4}	-	0	NA	-
Offsite SO_2 concentration (μg/m^3) - annual average[f]	1.01×10^{-5}	NA	2.47×10^{-5}	-	0	NA	-
Offsite gaseous fluorides (μg/m^3) 12-hour average[f]	0	NA	0	-	0	NA	-
Offsite gaseous fluorides (μg/m^3) 24-hour average[f]	0	NA	0	-	0	NA	-
Offsite gaseous fluorides (μg/m^3) 1-week average[f]	0	NA	0	-	0	NA	-
Offsite gaseous fluorides (μg/m^3) 1-month average[f]	0	NA	0	-	0	NA	-
Offsite HNO_3 concentration (μg/m^3) - 24-hour average[f]	2.54×10^{-3}	NA	7.74×10^{-3}	-	0	NA	-
Offsite HNO_3 concentration (μg/m^3) - annual average[f]	1.60×10^{-4}	NA	4.86×10^{-4}	-	0	NA	-
Onsite CO concentration (mg/m^3) - 8-hour average[f]	1.95×10^{-3}	NA	4.79×10^{-3}	-	0	NA	-
Onsite NO_x concentration (mg/m^3) - 1-hour average[f]	5.01×10^{-3}	NA	1.53×10^{-2}	-	0	NA	-
Onsite SO_2 concentration (mg/m^3) - 8-hour average[f]	2.84×10^{-6}	NA	6.93×10^{-6}	-	0	NA	-
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[f]	4.48×10^{-5}	NA	1.37×10^{-4}	-	0	NA	-
Onsite CO_2 concentration (mg/m^3) - 8-hour average[f]	1.42×10^{-6}	NA	3.47×10^{-6}	-	0	NA	-
Average number of radiation workers	78	NA	229	-	1,500	NA	-
Collective worker dose (person-rem/year)	5.76	NA	7.28	-	60	NA	-
Water usage (millions of liters/year)	32	NA	217	-	1,476	NA	-
Electricity usage (megawatt-hour/year)	5,262	NA	19,419	-	50,371	NA	-
Steam usage (millions of kilograms/year)	4	NA	88	-	167	NA	-
Fuel usage (thousands of liters/year)	17	NA	549	-	0	NA	-
High-level liquid waste generation (millions of liters per year)	0	NA	0.13	-	0	NA	-
Equivalent DWPF canisters (per year)	0	NA	1	-	405	NA	-
Saltstone generation (cubic meters/year)	0	NA	356	-	0	NA	-
TRU waste generation (cubic meters/year)	79	NA	79	-	1,136	NA	-
Hazardous/mixed waste generation (cubic meters/year)	96	NA	96	-	3,055	NA	-
Low-level waste generation (cubic meters/year)	617	NA	1,186	-	2,731	NA	-

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Atmospheric releases from the waste tank farms in 1993 contributed less than 1×10^{-8} rem to the maximally exposed individual and less than 50 person-rem to the worker collective dose. At this time, DOE does not expect any discernible increase to these or other environmental factors as a result of this phase in the stabilization alternative.

d. DOE does not expect any discernible incremental impacts as a result of this phase in the stabilization alternative.

e. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

f. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

Table D-37. Data for impact analysis of continuing storage (No Action) of plutonium and uranium stored in vaults (plutonium-238 scrap material).[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	4.0×10^{-7}	NA	NA	NA	NA	NA	NA
Liquid MEI dose (rem/year)	0	NA	NA	NA	NA	NA	NA
Total MEI dose (rem/year)	2.0×10^{-6}	NA	NA	NA	NA	NA	NA
Atmospheric population dose (person-rem/year)[c]	1.6×10^{-2}	NA	NA	NA	NA	NA	NA
Liquid population dose (person-rem/year)[c]	0	NA	NA	NA	NA	NA	NA
Total population dose (person-rem/year)	2.1×10^{-2}	NA	NA	NA	NA	NA	NA
Offsite CO concentration ($\mu g/m^3$) - 1-hour averaged[d]	1.22	NA	NA	NA	NA	NA	NA
Offsite CO concentration ($\mu g/m^3$) - 8-hour averaged[d]	0.292	NA	NA	NA	NA	NA	NA
Offsite NO_x concentration ($\mu g/m^3$) - annual averaged[d]	9.49×10^{-3}	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration ($\mu g/m^3$) - 3-hour averaged[d]	7.15×10^{-4}	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration ($\mu g/m^3$) - 24-hour averaged[d]	1.61×10^{-4}	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration ($\mu g/m^3$) - annual averaged[d]	1.01×10^{-5}	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 12-hour averaged[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 24-hour averaged[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 1-week averaged[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 1-month averaged[d]	0	NA	NA	NA	NA	NA	NA
Offsite HNO_3 concentration ($\mu g/m^3$) - 24-hour averaged[d]	2.54×10^{-3}	NA	NA	NA	NA	NA	NA
Offsite HNO_3 concentration ($\mu g/m^3$) - annual averaged[d]	1.60×10^{-4}	NA	NA	NA	NA	NA	NA
Onsite CO concentration (mg/m^3) - 8-hour averaged[d]	1.95×10^{-3}	NA	NA	NA	NA	NA	NA
Onsite NO_x concentration (mg/m^3) - 1-hour averaged[d]	5.01×10^{-3}	NA	NA	NA	NA	NA	NA
Onsite SO_2 concentration (mg/m^3) - 8-hour averaged[d]	2.48×10^{-6}	NA	NA	NA	NA	NA	NA
Onsite HNO_3 concentration (mg/m^3) - 8-hour averaged[d]	4.48×10^{-5}	NA	NA	NA	NA	NA	NA
Onsite CO_2 concentration (mg/m^3) - 8-hour averaged[d]	1.42×10^{-6}	NA	NA	NA	NA	NA	NA
Average number of radiation workers	78	NA	NA	NA	NA	NA	NA
Collective worker dose (person-rem/year)	5.76	NA	NA	NA	NA	NA	NA
Water usage (millions of liters/year)	32	NA	NA	NA	NA	NA	NA
Electricity usage (megawatt-hour/year)	5,262	NA	NA	NA	NA	NA	NA
Steam usage (millions of kilograms/year)	4	NA	NA	NA	NA	NA	NA
Fuel usage (thousands of liters/year)	17	NA	NA	NA	NA	NA	NA
High-level liquid waste generation (millions of liters per year)	0	NA	NA	NA	NA	NA	NA
Equivalent DWPF canisters (per year)	0	NA	NA	NA	NA	NA	NA
Saltstone generation (cubic meters/year)	0	NA	NA	NA	NA	NA	NA
TRU waste generation (cubic meters/year)	79	NA	NA	NA	NA	NA	NA
Hazardous/mixed waste generation (cubic meters/year)	96	NA	NA	NA	NA	NA	NA
Low-level waste generation (cubic meters/year)	617	NA	NA	NA	NA	NA	NA

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-38. Data for impact analysis of processing Mark-31 targets to metal.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	1.9×10^{-7}	NA	1.1×10^{-5}	2.3×10^{-9}	NA	6.6×10^{-9}	4.4×10^{-9}
Liquid MEI dose (rem/year)	0	NA	3.1×10^{-7}	2.2×10^{-10}	NA	7.3×10^{-10}	4.3×10^{-10}
Total MEI dose (rem/year)	1.9×10^{-7}	NA	1.1×10^{-5}	2.5×10^{-9}	NA	7.3×10^{-9}	4.8×10^{-9}
Atmospheric population dose (person-rem/year)[c]	1.2×10^{-2}	NA	0.49	1.0×10^{-4}	NA	2.9×10^{-4}	2.0×10^{-4}
Liquid population dose (person-rem/year)[c]	0	NA	1.8×10^{-3}	3.5×10^{-6}	NA	6.4×10^{-6}	6.8×10^{-6}
Total population dose (person-rem/year)	1.2×10^{-2}	NA	0.49	1.1×10^{-4}	NA	3.0×10^{-4}	2.0×10^{-4}
Offsite CO concentration (μg/m^3) - 1-hour average[d]	0	NA	0	0	NA	0	0
Offsite CO concentration (μg/m^3) - 8-hour average[d]	0	NA	0	0	NA	0	0
Offsite NO_x concentration (μg/m^3) - annual average[d]	0	NA	2.83×10^{-1}	0	NA	0	0
Offsite SO_2 concentration (μg/m^3) - 3-hour average[d]	0	NA	0	0	NA	0	0
Offsite SO_2 concentration (μg/m^3) - 24-hour average[d]	0	NA	0	0	NA	0	0
Offsite SO_2 concentration (μg/m^3) - annual average[d]	0	NA	0	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 12-hour average[d]	0	NA	5.24×10^{-2}	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 24-hour average[d]	0	NA	2.82×10^{-2}	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 1-week average[d]	0	NA	1.11×10^{-2}	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 1-month average[d]	0	NA	3.12×10^{-3}	0	NA	0	0
Offsite HNO_3 concentration (μg/m^3) - 24-hour average[d]	0	NA	0.777	0	NA	0	0
Offsite HNO_3 concentration (μg/m^3) - annual average[d]	0	NA	5.27×10^{-2}	0	NA	0	0
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	NA	0	0
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	0	NA	0.179	0	NA	0	0
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	NA	0	0
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	0	NA	1.22×10^{-2}	0	NA	0	0
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	NA	0	0
Average number of radiation workers	6	NA	719	47	NA	157	80
Collective worker dose (person-rem/year)	0.88	NA	160	2	NA	31	16
Water usage (millions of liters/year)	1	NA	1,500	278	NA	500	454
Electricity usage (megawatt-hour/year)	1	NA	27,722	3,683	NA	6,620	6,018
Steam usage (millions of kilograms/year)	1	NA	136	4	NA	7	6
Fuel usage (thousands of liters/year)	1	NA	888	12	NA	21	19
High-level liquid waste generation (millions of liters per year)	7.1×10^{-2}	NA	2.05	0	NA	0	0
Equivalent DWPF canisters (per year)	2	NA	42	0	NA	0	0
Saltstone generation (cubic meters/year)	193	NA	5,593	0	NA	0	0
TRU waste generation (cubic meters/year)	0	NA	64	2	NA	1	0
Hazardous/mixed waste generation (cubic meters/year)	3	NA	8	1	NA	1	0
Low-level waste generation (cubic meters/year)	1,797	NA	10,694	967	NA	7	5

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

Table D-39. Data for impact analysis of processing Mark-31 targets to oxide.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	1.9×10^{-7}	NA	1.1×10^{-5}	2.3×10^{-9}	NA	6.6×10^{-9}	4.4×10^{-9}
Liquid MEI dose (rem/year)	0	NA	3.1×10^{-7}	2.2×10^{-10}	NA	7.3×10^{-10}	4.3×10^{-10}
Total MEI dose (rem/year)	1.9×10^{-7}	NA	1.1×10^{-5}	2.5×10^{-9}	NA	7.3×10^{-9}	4.8×10^{-9}
Atmospheric population dose (person-rem/year)[c]	1.2×10^{-2}	NA	0.49	1.0×10^{-4}	NA	2.9×10^{-4}	2.0×10^{-4}
Liquid population dose (person-rem/year)[c]	0	NA	1.8×10^{-3}	3.5×10^{-6}	NA	6.4×10^{-6}	6.8×10^{-6}
Total population dose (person-rem/year)	1.2×10^{-2}	NA	0.49	1.1×10^{-4}	NA	3.0×10^{-4}	2.0×10^{-4}
Offsite CO concentration ($\mu g/m^3$) - 1-hour average[d]	0	NA	0	0	NA	0	0
Offsite CO concentration ($\mu g/m^3$) - 8-hour average[d]	0	NA	0	0	NA	0	0
Offsite NO_x concentration ($\mu g/m^3$) - annual average[d]	0	NA	0.283	0	NA	0	0
Offsite SO_2 concentration ($\mu g/m^3$) - 3-hour average[d]	0	NA	0	0	NA	0	0
Offsite SO_2 concentration ($\mu g/m^3$) - 24-hour average[d]	0	NA	0	0	NA	0	0
Offsite SO_2 concentration ($\mu g/m^3$) - annual average[d]	0	NA	0	0	NA	0	0
Offsite gaseous fluorides ($\mu g/m^3$) 12-hour average[d]	0	NA	5.24×10^{-2}	0	NA	0	0
Offsite gaseous fluorides ($\mu g/m^3$) 24-hour average[d]	0	NA	2.82×10^{-2}	0	NA	0	0
Offsite gaseous fluorides ($\mu g/m^3$) 1-week average[d]	0	NA	1.11×10^{-2}	0	NA	0	0
Offsite gaseous fluorides ($\mu g/m^3$) 1-month average[d]	0	NA	3.12×10^{-3}	0	NA	0	0
Offsite HNO_3 concentration ($\mu g/m^3$) - 24-hour average[d]	0	NA	0.777	0	NA	0	0
Offsite HNO_3 concentration ($\mu g/m^3$) - annual average[d]	0	NA	5.27×10^{-2}	0	NA	0	0
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	NA	0	0
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	0	NA	0.179	0	NA	0	0
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	NA	0	0
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	0	NA	1.22×10^{-2}	0	NA	0	0
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	NA	0	0
Average number of radiation workers	6	NA	719	47	NA	157	80
Collective worker dose (person-rem/year)	0.88	NA	160	2	NA	31	16
Water usage (millions of liters/year)	1	NA	1,500	278	NA	500	454
Electricity usage (megawatt-hour/year)	1	NA	27,722	3,683	NA	6,620	6,018
Steam usage (millions of kilograms/year)	1	NA	136	4	NA	7	6
Fuel usage (thousands of liters/year)	1	NA	888	12	NA	21	19
High-level liquid waste generation (millions of liters per year)	7.1×10^{-2}	NA	2.1	0	NA	0	0
Equivalent DWPF canisters (per year)	2	NA	42	0	NA	0	0
Saltstone generation (cubic meters/year)	193	NA	5,593	0	NA	0	0
TRU waste generation (cubic meters/year)	0	NA	64	2	NA	1	0
Hazardous/mixed waste generation (cubic meters/year)	3	NA	8	1	NA	1	0
Low-level waste generation (cubic meters/year)	1,797	NA	10,694	967	NA	7	5

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-40. Data for impact analysis of improving storage of Mark-31 targets.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	1.9×10^{-7}	1.9×10^{-7}	NA	1.9×10^{-7}	NA	2.8×10^{-10}	0
Liquid MEI dose (rem/year)	0	0	NA	0	NA	4.4×10^{-7}	0
Total MEI dose (rem/year)	1.9×10^{-7}	1.9×10^{-7}	NA	1.9×10^{-7}	NA	4.4×10^{-7}	0
Atmospheric population dose (person-rem/year)[c]	1.2×10^{-2}	1.2×10^{-2}	NA	1.2×10^{-2}	NA	1.3×10^{-5}	0
Liquid population dose (person-rem/year)[c]	0	0	NA	0	NA	2.6×10^{-3}	0
Total population dose (person-rem/year)	1.2×10^{-2}	1.2×10^{-2}	NA	1.2×10^{-2}	NA	2.6×10^{-3}	0
Offsite CO concentration (μg/m^3) - 1-hour average[d]	0	0	NA	0	NA	0	0
Offsite CO concentration (μg/m^3) - 8-hour average[d]	0	0	NA	0	NA	0	0
Offsite NO_x concentration (μg/m^3) - annual average[d]	0	0	NA	0	NA	0	0
Offsite SO_2 concentration (μg/m^3) - 3-hour average[d]	0	0	NA	0	NA	0	0
Offsite SO_2 concentration (μg/m^3) - 24-hour average[d]	0	0	NA	0	NA	0	0
Offsite SO_2 concentration (μg/m^3) - annual average[d]	0	0	NA	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 12-hour average[d]	0	0	NA	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 24-hour average[d]	0	0	NA	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 1-week average[d]	0	0	NA	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 1-month average[d]	0	0	NA	0	NA	0	0
Offsite HNO_3 concentration (μg/m^3) - 24-hour average[d]	0	0	NA	0	NA	0	0
Offsite HNO_3 concentration (μg/m^3) - annual average[d]	0	0	NA	0	NA	0	0
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	0	NA	0	NA	0	0
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	0	0	NA	0	NA	0	0
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	0	NA	0	NA	0	0
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	0	0	NA	0	NA	0	0
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	0	NA	0	NA	0	0
Average number of radiation workers	6	11	NA	6	NA	150	17
Collective worker dose (person-rem/year)	0.88	2.3	NA	0.88	NA	6	0.66
Water usage (millions of liters/year)	1	1	NA	1	NA	35	4
Electricity usage (megawatt-hour/year)	1	2	NA	1	NA	1,430	2
Steam usage (millions of kilograms/year)	1	1	NA	1	NA	11	0
Fuel usage (thousands of liters/year)	1	2	NA	1	NA	8	1
High-level liquid waste generation (millions of liters per year)	7.1×10^{-2}	0.19	NA	7.1×10^{-2}	NA	4.1×10^{-2}	0
Equivalent DWPF canisters (per year)	2	4	NA	2	NA	0	0
Saltstone generation (cubic meters/year)	193	516	NA	193	NA	11	0
TRU waste generation (cubic meters/year)	0	0	NA	0	NA	0	0
Hazardous/mixed waste generation (cubic meters/year)	3	8	NA	3	NA	0	0
Low-level waste generation (cubic meters/year)	1,797	4,664	NA	1,797	NA	128	92

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-41. Data for impact analysis of improving storage (accelerated schedule) of Mark-31 targets.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	1.9×10^{-7}	1.9×10^{-7}	NA	1.9×10^{-7}	NA	2.8×10^{-10}	0
Liquid MEI dose (rem/year)	0	0	NA	0	NA	4.4×10^{-7}	0
Total MEI dose (rem/year)	1.9×10^{-7}	1.9×10^{-7}	NA	1.9×10^{-7}	NA	4.4×10^{-7}	0
Atmospheric population dose (person-rem/year)[c]	1.2×10^{-2}	1.2×10^{-2}	NA	1.2×10^{-2}	NA	1.3×10^{-5}	0
Liquid population dose (person-rem/year)[c]	0	0	NA	0	NA	2.6×10^{-3}	0
Total population dose (person-rem/year)	1.2×10^{-2}	1.2×10^{-2}	NA	1.2×10^{-2}	NA	2.6×10^{-3}	0
Offsite CO concentration ($\mu g/m^3$) - 1-hour average[d]	0	0	NA	0	NA	0	0
Offsite CO concentration ($\mu g/m^3$) - 8-hour average[d]	0	0	NA	0	NA	0	0
Offsite NO_x concentration ($\mu g/m^3$) - annual average[d]	0	0	NA	0	NA	0	0
Offsite SO_2 concentration ($\mu g/m^3$) - 3-hour average[d]	0	0	NA	0	NA	0	0
Offsite SO_2 concentration ($\mu g/m^3$) - 24-hour average[d]	0	0	NA	0	NA	0	0
Offsite SO_2 concentration ($\mu g/m^3$) - annual average[d]	0	0	NA	0	NA	0	0
Offsite gaseous fluorides ($\mu g/m^3$) 12-hour average[d]	0	0	NA	0	NA	0	0
Offsite gaseous fluorides ($\mu g/m^3$) 24-hour average[d]	0	0	NA	0	NA	0	0
Offsite gaseous fluorides ($\mu g/m^3$) 1-week average[d]	0	0	NA	0	NA	0	0
Offsite gaseous fluorides ($\mu g/m^3$) 1-month average[d]	0	0	NA	0	NA	0	0
Offsite HNO_3 concentration ($\mu g/m^3$) - 24-hour average[d]	0	0	NA	0	NA	0	0
Offsite HNO_3 concentration ($\mu g/m^3$) - annual average[d]	0	0	NA	0	NA	0	0
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	0	NA	0	NA	0	0
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	0	0	NA	0	NA	0	0
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	0	NA	0	NA	0	0
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	0	0	NA	0	NA	0	0
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	0	NA	0	NA	0	0
Average number of radiation workers	6	11	NA	6	NA	150	17
Collective worker dose (person-rem/year)	0.88	2.3	NA	0.88	NA	6	0.66
Water usage (millions of liters/year)	1	1	NA	1	NA	35	4
Electricity usage (megawatt-hour/year)	1	2	NA	1	NA	1,430	2
Steam usage (millions of kilograms/year)	1	1	NA	1	NA	11	0
Fuel usage (thousands of liters/year)	1	2	NA	1	NA	8	1
High-level liquid waste generation (millions of liters per year)	7.1×10^{-2}	0.19	NA	7.1×10^{-2}	NA	4.1×10^{-2}	0
Equivalent DWPF canisters (per year)	2	4	NA	2	NA	0	0
Saltstone generation (cubic meters/year)	193	516	NA	193	NA	11	0
TRU waste generation (cubic meters/year)	0	0	NA	0	NA	0	0
Hazardous/mixed waste generation (cubic meters/year)	3	8	NA	3	NA	0	0
Low-level waste generation (cubic meters/year)	1,797	4,664	NA	1,797	NA	128	92

TC

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

Table D-42. Data for impact analysis of processing and storage for vitrification (DWPF) of Mark-31 targets.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage[c]	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage[d]
Atmospheric MEI dose (rem/year)	1.9×10^{-7}	NA	8.7×10^{-6}	-	1.6×10^{-4}	NA	-
Liquid MEI dose (rem/year)	0	NA	2.5×10^{-7}	-	2.4×10^{-5}	NA	-
Total MEI dose (rem/year)	1.9×10^{-7}	NA	9.0×10^{-6}	-	1.8×10^{-4}	NA	-
Atmospheric population dose (person-rem/year)[e]	1.2×10^{-2}	NA	0.39	-	6.9	NA	-
Liquid population dose (person-rem/year)[e]	0	NA	1.5×10^{-3}	-	0.14	NA	-
Total population dose (person-rem/year)	1.2×10^{-2}	NA	0.39	-	7.0	NA	-
Offsite CO concentration (μg/m³) - 1-hour average[f]	0	NA	0	-	0	NA	-
Offsite CO concentration (μg/m³) - 8-hour average[f]	0	NA	0	-	0	NA	-
Offsite NO_x concentration (μg/m³) - annual average[f]	0	NA	0.227	-	0	NA	-
Offsite SO_2 concentration (μg/m³) - 3-hour average[f]	0	NA	0	-	0	NA	-
Offsite SO_2 concentration (μg/m³) - 24-hour average[f]	0	NA	0	-	0	NA	-
Offsite SO_2 concentration (μg/m³) - annual average[f]	0	NA	0	-	0	NA	-
Offsite gaseous fluorides (μg/m³) 12-hour average[f]	0	NA	4.19×10^{-2}	-	0	NA	-
Offsite gaseous fluorides (μg/m³) 24-hour average[f]	0	NA	2.26×10^{-2}	-	0	NA	-
Offsite gaseous fluorides (μg/m³) 1-week average[f]	0	NA	8.87×10^{-3}	-	0	NA	-
Offsite gaseous fluorides (μg/m³) 1-month average[f]	0	NA	2.50×10^{-3}	-	0	NA	-
Offsite HNO_3 concentration (μg/m³) - 24-hour average[f]	0	NA	0.622	-	0	NA	-
Offsite HNO_3 concentration (μg/m³) - annual average[f]	0	NA	4.22×10^{-2}	-	0	NA	-
Onsite CO concentration (mg/m³) - 8-hour average[f]	0	NA	0	-	0	NA	-
Onsite NO_x concentration (mg/m³) - 1-hour average[f]	0	NA	0.143	-	0	NA	-
Onsite SO_2 concentration (mg/m³) - 8-hour average[f]	0	NA	0	-	0	NA	-
Onsite HNO_3 concentration (mg/m³) - 8-hour average[f]	0	NA	9.77×10^{-3}	-	0	NA	-
Onsite CO_2 concentration (mg/m³) - 8-hour average[f]	0	NA	0	-	0	NA	-
Average number of radiation workers	6	NA	338	-	1,500	NA	-
Collective worker dose (person-rem/year)	0.88	NA	54	-	60	NA	-
Water usage (millions of liters/year)	1	NA	1,200	-	1,476	NA	-
Electricity usage (megawatt-hour/year)	1	NA	22,178	-	50,371	NA	-
Steam usage (millions of kilograms/year)	1	NA	109	-	167	NA	-
Fuel usage (thousands of liters/year)	1	NA	710	-	0	NA	-
High-level liquid waste generation (millions of liters per year)	7.1×10^{-2}	NA	1.6	-	0	NA	-
Equivalent DWPF canisters (per year)	2	NA	33	-	405	NA	-
Saltstone generation (cubic meters/year)	193	NA	4,404	-	0	NA	-
TRU waste generation (cubic meters/year)	0	NA	0	-	1,136	NA	-
Hazardous/mixed waste generation (cubic meters/year)	3	NA	8	-	3,055	NA	-
Low-level waste generation (cubic meters/year)	1,797	NA	5,824	-	2,731	NA	-

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Atmospheric releases from the waste tank farms in 1993 contributed less than 1×10^{-8} rem to the maximally exposed individual and less than 50 person-rem to the worker collective dose. At this time, DOE does not expect any discernible increase to these or other environmental factors as a result of this phase in the stabilization alternative.

d. DOE does not expect any discernible incremental impacts as a result of this phase in the stabilization alternative.

e. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

f. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-43. Data for impact analysis of vitrification (F-Canyon) of Mark-31 targets.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	1.9×10^{-7}	NA	1.1×10^{-5}	2.3×10^{-9}	1.3×10^{-5}	NA	4.4×10^{-9}
Liquid MEI dose (rem/year)	0	NA	3.1×10^{-7}	2.2×10^{-10}	3.8×10^{-7}	NA	4.3×10^{-10}
Total MEI dose (rem/year)	1.9×10^{-7}	NA	1.1×10^{-5}	2.5×10^{-9}	1.3×10^{-5}	NA	4.8×10^{-9}
Atmospheric population dose (person-rem/year)[c]	1.2×10^{-2}	NA	0.49	1.0×10^{-4}	0.57	NA	2.0×10^{-4}
Liquid population dose (person-rem/year)[c]	0	NA	1.8×10^{-3}	3.5×10^{-6}	2.2×10^{-3}	NA	6.8×10^{-6}
Total population dose (person-rem/year)	1.2×10^{-2}	NA	0.49	1.1×10^{-4}	0.57	NA	2.0×10^{-4}
Offsite CO concentration ($\mu g/m^3$) - 1-hour average[d]	0	NA	0	0	0	NA	0
Offsite CO concentration ($\mu g/m^3$) - 8-hour average[d]	0	NA	0	0	0	NA	0
Offsite NO_x concentration ($\mu g/m^3$) - annual average[d]	0	NA	0.283	0	0.340	NA	0
Offsite SO_2 concentration ($\mu g/m^3$) - 3-hour average[d]	0	NA	0	0	0	NA	0
Offsite SO_2 concentration ($\mu g/m^3$) - 24-hour average[d]	0	NA	0	0	0	NA	0
Offsite SO_2 concentration ($\mu g/m^3$) - annual average[d]	0	NA	0	0	0	NA	0
Offsite gaseous fluorides ($\mu g/m^3$) 12-hour average[d]	0	NA	5.24×10^{-2}	0	6.29×10^{-2}	NA	0
Offsite gaseous fluorides ($\mu g/m^3$) 24-hour average[d]	0	NA	2.82×10^{-2}	0	3.39×10^{-2}	NA	0
Offsite gaseous fluorides ($\mu g/m^3$) 1-week average[d]	0	NA	1.11×10^{-2}	0	1.33×10^{-2}	NA	0
Offsite gaseous fluorides ($\mu g/m^3$) 1-month average[d]	0	NA	3.12×10^{-3}	0	3.75×10^{-3}	NA	0
Offsite HNO_3 concentration ($\mu g/m^3$) - 24-hour average[d]	0	NA	0.777	0	0.933	NA	0
Offsite HNO_3 concentration ($\mu g/m^3$) - annual average[d]	0	NA	5.27×10^{-2}	0	6.33×10^{-2}	NA	0
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	0	NA	0
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	0	NA	0.179	0	0.215	NA	0
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	0	NA	0
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	0	NA	1.22×10^{-2}	0	1.47×10^{-2}	NA	0
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	0	NA	0
Average number of radiation workers	6	NA	719	47	790	NA	80
Collective worker dose (person-rem/year)	0.88	NA	160	2	171	NA	16
Water usage (millions of liters/year)	1	NA	1,500	278	1,799	NA	454
Electricity usage (megawatt-hour/year)	1	NA	27,722	3,683	33,264	NA	6,018
Steam usage (millions of kilograms/year)	1	NA	136	4	163	NA	6
Fuel usage (thousands of liters/year)	1	NA	888	12	1,063	NA	19
High-level liquid waste generation (millions of liters per year)	7.1×10^{-2}	NA	2.1	0	1.9	NA	0
Equivalent DWPF canisters (per year)	2	NA	42	0	38	NA	0
Saltstone generation (cubic meters/year)	193	NA	5,593	0	5,077	NA	0
TRU waste generation (cubic meters/year)	0	NA	64	2	64	NA	0
Hazardous/mixed waste generation (cubic meters/year)	3	NA	8	1	0	NA	0
Low-level waste generation (cubic meters/year)	1,797	NA	10,694	967	6,146	NA	5

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-44. Data for impact analysis of continuing storage (No Action) of Mark-31 targets.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	1.9×10^{-7}	1.9×10^{-7}	NA	1.9×10^{-7}	NA	NA	NA
Liquid MEI dose (rem/year)	0	0	NA	0	NA	NA	NA
Total MEI dose (rem/year)	1.9×10^{-7}	1.9×10^{-7}	NA	1.9×10^{-7}	NA	NA	NA
Atmospheric population dose (person-rem/year)[c]	1.2×10^{-2}	1.2×10^{-2}	NA	1.2×10^{-2}	NA	NA	NA
Liquid population dose (person-rem/year)[c]	0	0	NA	0	NA	NA	NA
Total population dose (person-rem/year)	1.2×10^{-2}	1.2×10^{-2}	NA	1.2×10^{-2}	NA	NA	NA
Offsite CO concentration ($\mu g/m^3$) - 1-hour average[d]	0	0	NA	0	NA	NA	NA
Offsite CO concentration ($\mu g/m^3$) - 8-hour average[d]	0	0	NA	0	NA	NA	NA
Offsite NO_x concentration ($\mu g/m^3$) - annual average[d]	0	0	NA	0	NA	NA	NA
Offsite SO_2 concentration ($\mu g/m^3$) - 3-hour average[d]	0	0	NA	0	NA	NA	NA
Offsite SO_2 concentration ($\mu g/m^3$) - 24-hour average[d]	0	0	NA	0	NA	NA	NA
Offsite SO_2 concentration ($\mu g/m^3$) - annual average[d]	0	0	NA	0	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 12-hour average[d]	0	0	NA	0	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 24-hour average[d]	0	0	NA	0	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 1-week average[d]	0	0	NA	0	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 1-month average[d]	0	0	NA	0	NA	NA	NA
Offsite HNO_3 concentration ($\mu g/m^3$) - 24-hour average[d]	0	0	NA	0	NA	NA	NA
Offsite HNO_3 concentration ($\mu g/m^3$) - annual average[d]	0	0	NA	0	NA	NA	NA
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	0	NA	0	NA	NA	NA
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	0	0	NA	0	NA	NA	NA
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	0	NA	0	NA	NA	NA
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	0	0	NA	0	NA	NA	NA
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	0	NA	0	NA	NA	NA
Average number of radiation workers	6	11	NA	6	NA	NA	NA
Collective worker dose (person-rem/year)	0.88	2.3	NA	0.88	NA	NA	NA
Water usage (millions of liters/year)	1	1	NA	1	NA	NA	NA
Electricity usage (megawatt-hour/year)	1	2	NA	1	NA	NA	NA
Steam usage (millions of kilograms/year)	1	1	NA	1	NA	NA	NA
Fuel usage (thousands of liters/year)	1	2	NA	1	NA	NA	NA
High-level liquid waste generation (millions of liters per year)	7.1×10^{-2}	0.19	NA	7.1×10^{-2}	NA	NA	NA
Equivalent DWPF canisters (per year)	2	4	NA	2	NA	NA	NA
Saltstone generation (cubic meters/year)	193	516	NA	193	NA	NA	NA
TRU waste generation (cubic meters/year)	0	0	NA	0	NA	NA	NA
Hazardous/mixed waste generation (cubic meters/year)	3	8	NA	3	NA	NA	NA
Low-level waste generation (cubic meters/year)	1,797	4,664	NA	1,797	NA	NA	NA

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

Table D-45. Data for impact analysis of blending Mark 16/22 fuels to low enriched uranium.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage[a]
Atmospheric MEI dose (rem/year)	5.0×10^{-8}	NA	6.6×10^{-4}	1.4×10^{-7}	8.5×10^{-7}	NA	-
Liquid MEI dose (rem/year)	0	NA	9.8×10^{-6}	5.7×10^{-7}	2.5×10^{-8}	NA	-
Total MEI dose (rem/year)	5.0×10^{-8}	NA	6.7×10^{-4}	7.1×10^{-7}	8.8×10^{-7}	NA	-
Atmospheric population dose (person-rem/year)[c]	3.2×10^{-3}	NA	27	5.8×10^{-3}	3.8×10^{-2}	NA	-
Liquid population dose (person-rem/year)[c]	0	NA	3.3×10^{-2}	1.6×10^{-3}	1.5×10^{-4}	NA	-
Total population dose (person-rem/year)	3.2×10^{-3}	NA	27	7.5×10^{-3}	3.8×10^{-2}	NA	-
Offsite CO concentration (μg/m^3) - 1-hour average[d]	0	NA	10.6	0.394	0	NA	-
Offsite CO concentration (μg/m^3) - 8-hour average[d]	0	NA	2.54	9.41×10^{-2}	0	NA	-
Offsite NO_x concentration (μg/m^3) - annual average[d]	0	NA	8.25×10^{-2}	5.28×10^{-3}	2.27×10^{-2}	NA	-
Offsite SO_2 concentration (μg/m^3) - 3-hour average[d]	0	NA	6.21×10^{-3}	2.25×10^{-4}	0	NA	-
Offsite SO_2 concentration (μg/m^3) - 24-hour average[d]	0	NA	1.40×10^{-3}	5.06×10^{-5}	0	NA	-
Offsite SO_2 concentration (μg/m^3) - annual average[d]	0	NA	8.77×10^{-5}	3.18×10^{-6}	0	NA	-
Offsite gaseous fluorides (μg/m^3) 12-hour average[d]	0	NA	7.48×10^{-6}	0	4.20×10^{-3}	NA	-
Offsite gaseous fluorides (μg/m^3) 24-hour average[d]	0	NA	2.98×10^{-6}	0	2.26×10^{-3}	NA	-
Offsite gaseous fluorides (μg/m^3) 1-week average[d]	0	NA	1.58×10^{-6}	0	8.88×10^{-4}	NA	-
Offsite gaseous fluorides (μg/m^3) 1-month average[d]	0	NA	4.46×10^{-7}	0	2.50×10^{-4}	NA	-
Offsite HNO_3 concentration (μg/m^3) - 24-hour average[d]	0	NA	2.21×10^{-2}	1.41×10^{-3}	6.22×10^{-2}	NA	-
Offsite HNO_3 concentration (μg/m^3) - annual average[d]	0	NA	1.39×10^{-3}	8.88×10^{-5}	4.22×10^{-3}	NA	-
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	1.70×10^{-2}	6.28×10^{-4}	0	NA	-
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	0	NA	4.36×10^{-2}	2.79×10^{-3}	1.43×10^{-2}	NA	-
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	2.46×10^{-5}	8.93×10^{-7}	0	NA	-
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	0	NA	3.90×10^{-4}	2.49×10^{-5}	9.77×10^{-4}	NA	-
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	1.22×10^{-5}	4.73×10^{-7}	0	NA	-
Average number of radiation workers	5	NA	362	23	33	NA	-
Collective worker dose (person-rem/year)	0.71	NA	16	0.23	5.1	NA	-
Water usage (millions of liters/year)	0	NA	309	24	120	NA	-
Electricity usage (megawatt-hour/year)	1	NA	23,597	1,802	2,218	NA	-
Steam usage (millions of kilograms/year)	1	NA	141	11	11	NA	-
Fuel usage (thousands of liters/year)	1	NA	888	68	71	NA	-
High-level liquid waste generation (millions of liters per year)	5.7×10^{-2}	NA	2.2	1.8×10^{-2}	0.14	NA	-
Equivalent DWPF canisters (per year)	1	NA	19	0	3	NA	-
Saltstone generation (cubic meters/year)	155	NA	6,086	50	388	NA	-
TRU waste generation (cubic meters/year)	0	NA	0	0	0	NA	-
Hazardous/mixed waste generation (cubic meters/year)	2	NA	8	0	0	NA	-
Low-level waste generation (cubic meters/year)	1,453	NA	5,693	63	116	NA	-

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

Table D-46. Data for impact analysis of processing Mark 16/22 fuels to oxide.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	5.0×10^{-8}	NA	6.6×10^{-4}	1.4×10^{-7}	1.0×10^{-11}	NA	2.1×10^{-12}
Liquid MEI dose (rem/year)	0	NA	9.8×10^{-6}	5.7×10^{-7}	0	NA	0
Total MEI dose (rem/year)	5.0×10^{-8}	NA	6.7×10^{-4}	7.1×10^{-7}	1.0×10^{-11}	NA	2.1×10^{-12}
Atmospheric population dose (person-rem/year)[c]	3.2×10^{-3}	NA	27	5.8×10^{-3}	4.2×10^{-7}	NA	8.4×10^{-8}
Liquid population dose (person-rem/year)[c]	0	NA	3.3×10^{-2}	1.6×10^{-3}	0	NA	0
Total population dose (person-rem/year)	3.2×10^{-3}	NA	27	7.5×10^{-3}	4.2×10^{-7}	NA	8.4×10^{-8}
Offsite CO concentration ($\mu g/m^3$) - 1-hour average[d]	0	NA	10.6	0.394	0	NA	0
Offsite CO concentration ($\mu g/m^3$) - 8-hour average[d]	0	NA	2.54	9.41×10^{-2}	0	NA	0
Offsite NO_x concentration ($\mu g/m^3$) - annual average[d]	0	NA	8.25×10^{-2}	5.28×10^{-3}	2.82×10^{-3}	NA	5.63×10^{-4}
Offsite SO_2 concentration ($\mu g/m^3$) - 3-hour average[d]	0	NA	6.21×10^{-3}	2.25×10^{-4}	0	NA	0
Offsite SO_2 concentration ($\mu g/m^3$) - 24-hour average[d]	0	NA	1.40×10^{-3}	5.06×10^{-5}	0	NA	0
Offsite SO_2 concentration ($\mu g/m^3$) - annual average[d]	0	NA	8.77×10^{-5}	3.18×10^{-6}	0	NA	0
Offsite gaseous fluorides ($\mu g/m^3$) 12-hour average[d]	0	NA	7.48×10^{-6}	0	0	NA	0
Offsite gaseous fluorides ($\mu g/m^3$) 24-hour average[d]	0	NA	2.98×10^{-6}	0	0	NA	0
Offsite gaseous fluorides ($\mu g/m^3$) 1-week average[d]	0	NA	1.58×10^{-6}	0	0	NA	0
Offsite gaseous fluorides ($\mu g/m^3$) 1-month average[d]	0	NA	4.46×10^{-7}	0	0	NA	0
Offsite HNO_3 concentration ($\mu g/m^3$) - 24-hour average[d]	0	NA	2.21×10^{-2}	1.41×10^{-3}	0	NA	0
Offsite HNO_3 concentration ($\mu g/m^3$) - annual average[d]	0	NA	1.39×10^{-3}	8.88×10^{-5}	0	NA	0
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	1.70×10^{-2}	6.28×10^{-4}	0	NA	0
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	0	NA	4.36×10^{-2}	2.79×10^{-3}	1.49×10^{-3}	NA	2.98×10^{-4}
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	2.46×10^{-5}	8.93×10^{-7}	0	NA	0
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	0	NA	3.90×10^{-4}	2.49×10^{-5}	0	NA	0
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	1.22×10^{-5}	4.73×10^{-7}	0	NA	0
Average number of radiation workers	5	NA	362	23	80	NA	16
Collective worker dose (person-rem/year)	0.71	NA	16	0.23	51	NA	0.16
Water usage (millions of liters/year)	0	NA	309	24	70	NA	14
Electricity usage (megawatt-hour/year)	1	NA	23,597	1,802	5,363	NA	1,073
Steam usage (millions of kilograms/year)	1	NA	141	11	32	NA	6
Fuel usage (thousands of liters/year)	1	NA	888	68	201	NA	40
High-level liquid waste generation (millions of liters per year)	5.7×10^{-2}	NA	2.2	1.8×10^{-2}	0	NA	0
Equivalent DWPF canisters (per year)	1	NA	19	0	0	NA	0
Saltstone generation (cubic meters/year)	155	NA	6,086	50	0	NA	0
TRU waste generation (cubic meters/year)	0	NA	0	0	0	NA	0
Hazardous/mixed waste generation (cubic meters/year)	2	NA	8	0	0	NA	0
Low-level waste generation (cubic meters/year)	1,453	NA	5,693	63	103	NA	21

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-47. Data for impact analysis of improving storage of Mark 16/22 fuels.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	5.0×10^{-8}	NA	NA	5.0×10^{-8}	NA	2.8×10^{-10}	0
Liquid MEI dose (rem/year)	0	NA	NA	0	NA	4.4×10^{-7}	0
Total MEI dose (rem/year)	5.0×10^{-8}	NA	NA	5.0×10^{-8}	NA	4.4×10^{-7}	0
Atmospheric population dose (person-rem/year)[c]	3.2×10^{-3}	NA	NA	3.2×10^{-3}	NA	1.3×10^{-5}	0
Liquid population dose (person-rem/year)[c]	0	NA	NA	0	NA	2.6×10^{-3}	0
Total population dose (person-rem/year)	3.2×10^{-3}	NA	NA	3.2×10^{-3}	NA	2.6×10^{-3}	0
Offsite CO concentration ($\mu g/m^3$) - 1-hour average[d]	0	NA	NA	0	NA	0	0
Offsite CO concentration ($\mu g/m^3$) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Offsite NO_x concentration ($\mu g/m^3$) - annual average[d]	0	NA	NA	0	NA	0	0
Offsite SO_2 concentration ($\mu g/m^3$) - 3-hour average[d]	0	NA	NA	0	NA	0	0
Offsite SO_2 concentration ($\mu g/m^3$) - 24-hour average[d]	0	NA	NA	0	NA	0	0
Offsite SO_2 concentration ($\mu g/m^3$) - annual average[d]	0	NA	NA	0	NA	0	0
Offsite gaseous fluorides ($\mu g/m^3$) 12-hour average[d]	0	NA	NA	0	NA	0	0
Offsite gaseous fluorides ($\mu g/m^3$) 24-hour average[d]	0	NA	NA	0	NA	0	0
Offsite gaseous fluorides ($\mu g/m^3$) 1-week average[d]	0	NA	NA	0	NA	0	0
Offsite gaseous fluorides ($\mu g/m^3$) 1-month average[d]	0	NA	NA	0	NA	0	0
Offsite HNO_3 concentration ($\mu g/m^3$) - 24-hour average[d]	0	NA	NA	0	NA	0	0
Offsite HNO_3 concentration ($\mu g/m^3$) - annual average[d]	0	NA	NA	0	NA	0	0
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	0	NA	NA	0	NA	0	0
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Average number of radiation workers	5	NA	NA	5	NA	150	13
Collective worker dose (person-rem/year)	0.71	NA	NA	0.71	NA	6	0.53
Water usage (millions of liters/year)	0	NA	NA	0	NA	35	3
Electricity usage (megawatt-hour/year)	1	NA	NA	1	NA	1,430	1
Steam usage (millions of kilograms/year)	1	NA	NA	1	NA	11	0
Fuel usage (thousands of liters/year)	1	NA	NA	1	NA	8	0
High-level liquid waste generation (millions of liters per year)	5.7×10^{-2}	NA	NA	5.7×10^{-2}	NA	4.1×10^{-2}	0
Equivalent DWPF canisters (per year)	1	NA	NA	1	NA	0	0
Saltstone generation (cubic meters/year)	155	NA	NA	155	NA	11	0
TRU waste generation (cubic meters/year)	0	NA	NA	0	NA	0	0
Hazardous/mixed waste generation (cubic meters/year)	2	NA	NA	2	NA	0	0
Low-level waste generation (cubic meters/year)	1,453	NA	NA	1,453	NA	128	73

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

Table D-48. Data for impact analysis of improving storage (accelerated schedule) of Mark 16/22 fuels.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	5.0×10^{-8}	NA	NA	5.0×10^{-8}	NA	2.8×10^{-10}	0
Liquid MEI dose (rem/year)	0	NA	NA	0	NA	4.4×10^{-7}	0
Total MEI dose (rem/year)	5.0×10^{-8}	NA	NA	5.0×10^{-8}	NA	4.4×10^{-7}	0
Atmospheric population dose (person-rem/year)[c]	3.2×10^{-3}	NA	NA	3.2×10^{-3}	NA	1.3×10^{-5}	0
Liquid population dose (person-rem/year)[c]	0	NA	NA	0	NA	2.6×10^{-3}	0
Total population dose (person-rem/year)	3.2×10^{-3}	NA	NA	3.2×10^{-3}	NA	2.6×10^{-3}	0
Offsite CO concentration (μg/m^3) - 1-hour average[d]	0	NA	NA	0	NA	0	0
Offsite CO concentration (μg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Offsite NO_x concentration (μg/m^3) - annual average[d]	0	NA	NA	0	NA	0	0
Offsite SO_2 concentration (μg/m^3) - 3-hour average[d]	0	NA	NA	0	NA	0	0
Offsite SO_2 concentration (μg/m^3) - 24-hour average[d]	0	NA	NA	0	NA	0	0
Offsite SO_2 concentration (μg/m^3) - annual average[d]	0	NA	NA	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 12-hour average[d]	0	NA	NA	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 24-hour average[d]	0	NA	NA	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 1-week average[d]	0	NA	NA	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 1-month average[d]	0	NA	NA	0	NA	0	0
Offsite HNO_3 concentration (μg/m^3) - 24-hour average[d]	0	NA	NA	0	NA	0	0
Offsite HNO_3 concentration (μg/m^3) - annual average[d]	0	NA	NA	0	NA	0	0
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	0	NA	NA	0	NA	0	0
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Average number of radiation workers	5	NA	NA	5	NA	150	13
Collective worker dose (person-rem/year)	0.71	NA	NA	0.71	NA	6	0.53
Water usage (millions of liters/year)	0	NA	NA	0	NA	35	3
Electricity usage (megawatt-hour/year)	1	NA	NA	1	NA	1,430	1
Steam usage (millions of kilograms/year)	1	NA	NA	1	NA	11	0
Fuel usage (thousands of liters/year)	1	NA	NA	1	NA	8	0
High-level liquid waste generation (millions of liters per year)	5.7×10^{-2}	NA	NA	5.7×10^{-2}	NA	4.1×10^{-2}	0
Equivalent DWPF canisters (per year)	1	NA	NA	1	NA	0	0
Saltstone generation (cubic meters/year)	155	NA	NA	155	NA	11	0
TRU waste generation (cubic meters/year)	0	NA	NA	0	NA	0	0
Hazardous/mixed waste generation (cubic meters/year)	2	NA	NA	2	NA	0	0
Low-level waste generation (cubic meters/year)	1,453	NA	NA	1,453	NA	128	73

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

Table D-49. Data for impact analysis of processing and storage for vitrification (DWPF) of Mark 16/22 fuels.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage[c]	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage[d]
Atmospheric MEI dose (rem/year)	5.0×10^{-8}	NA	8.6×10^{-6}	-	1.6×10^{-4}	NA	-
Liquid MEI dose (rem/year)	0	NA	2.5×10^{-7}	-	2.4×10^{-8}	NA	-
Total MEI dose (rem/year)	5.0×10^{-8}	NA	8.8×10^{-6}	-	1.8×10^{-4}	NA	-
Atmospheric population dose (person-rem/year)[e]	3.2×10^{-3}	NA	0.38	-	6.9	NA	-
Liquid population dose (person-rem/year)[e]	0	NA	1.5×10^{-3}	-	0.14	NA	-
Total population dose (person-rem/year)	3.2×10^{-3}	NA	0.38	-	7.0	NA	-
Offsite CO concentration ($\mu g/m^3$) - 1-hour average[f]	0	NA	0	-	0	NA	-
Offsite CO concentration ($\mu g/m^3$) - 8-hour average[f]	0	NA	0	-	0	NA	-
Offsite NO_x concentration ($\mu g/m^3$) - annual average[f]	0	NA	0.227	-	0	NA	-
Offsite SO_2 concentration ($\mu g/m^3$) - 3-hour average[f]	0	NA	0	-	0	NA	-
Offsite SO_2 concentration ($\mu g/m^3$) - 24-hour average[f]	0	NA	0	-	0	NA	-
Offsite SO_2 concentration ($\mu g/m^3$) - annual average[f]	0	NA	0	-	0	NA	-
Offsite gaseous fluorides ($\mu g/m^3$) 12-hour average[f]	0	NA	4.19×10^{-2}	-	0	NA	-
Offsite gaseous fluorides ($\mu g/m^3$) 24-hour average[f]	0	NA	2.26×10^{-2}	-	0	NA	-
Offsite gaseous fluorides ($\mu g/m^3$) 1-week average[f]	0	NA	8.87×10^{-3}	-	0	NA	-
Offsite gaseous fluorides ($\mu g/m^3$) 1-month average[f]	0	NA	2.50×10^{-3}	-	0	NA	-
Offsite HNO_3 concentration ($\mu g/m^3$) - 24-hour average[f]	0	NA	0.622	-	0	NA	-
Offsite HNO_3 concentration ($\mu g/m^3$) - annual average[f]	0	NA	4.22×10^{-2}	-	0	NA	-
Onsite CO concentration (mg/m^3) - 8-hour average[f]	0	NA	0	-	0	NA	-
Onsite NO_x concentration (mg/m^3) - 1-hour average[f]	0	NA	0.143	-	0	NA	-
Onsite SO_2 concentration (mg/m^3) - 8-hour average[f]	0	NA	0	-	0	NA	-
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[f]	0	NA	9.77×10^{-3}	-	0	NA	-
Onsite CO_2 concentration (mg/m^3) - 8-hour average[f]	0	NA	0	-	0	NA	-
Average number of radiation workers	5	NA	338	-	1,500	NA	-
Collective worker dose (person-rem/year)	0.71	NA	54	-	60	NA	-
Water usage (millions of liters/year)	0	NA	1,200	-	1,476	NA	-
Electricity usage (megawatt-hour/year)	1	NA	22,178	-	50,371	NA	-
Steam usage (millions of kilograms/year)	1	NA	109	-	167	NA	-
Fuel usage (thousands of liters/year)	1	NA	710	-	0	NA	-
High-level liquid waste generation (millions of liters per year)	5.7×10^{-2}	NA	1.6	-	0	NA	-
Equivalent DWPF canisters (per year)	1	NA	33	-	405	NA	-
Saltstone generation (cubic meters/year)	155	NA	4,404	-	0	NA	-
TRU waste generation (cubic meters/year)	0	NA	0	-	1,136	NA	-
Hazardous/mixed waste generation (cubic meters/year)	2	NA	8	-	3,055	NA	-
Low-level waste generation (cubic meters/year)	1,453	NA	5,824	-	2,731	NA	-

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Atmospheric releases from the waste tank farms in 1993 contributed less than 1×10^{-8} rem to the maximally exposed individual and less than 50 person-rem to the worker collective dose. At this time, DOE does not expect any discernible increase to these or other environmental factors as a result of this phase in the stabilization alternative.

d. DOE does not expect any discernible incremental impacts as a result of this phase in the stabilization alternative.

e. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

f. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-50. Data for impact analysis of continuing storage (No Action) of Mark 16/22 fuels.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	5.0×10^{-8}	NA	NA	NA	NA	NA	NA
Liquid MEI dose (rem/year)	0	NA	NA	NA	NA	NA	NA
Total MEI dose (rem/year)	5.0×10^{-8}	NA	NA	NA	NA	NA	NA
Atmospheric population dose (person-rem/year)[c]	3.2×10^{-3}	NA	NA	NA	NA	NA	NA
Liquid population dose (person-rem/year)[c]	0	NA	NA	NA	NA	NA	NA
Total population dose (person-rem/year)	3.2×10^{-3}	NA	NA	NA	NA	NA	NA
Offsite CO concentration (μg/m^3) - 1-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite CO concentration (μg/m^3) - 8-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite NO_x concentration (μg/m^3) - annual average[d]	0	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration (μg/m^3) - 3-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration (μg/m^3) - 24-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration (μg/m^3) - annual average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 12-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 24-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 1-week average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (μg/m^3) 1-month average[d]	0	NA	NA	NA	NA	NA	NA
Offsite HNO_3 concentration (μg/m^3) - 24-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite HNO_3 concentration (μg/m^3) - annual average[d]	0	NA	NA	NA	NA	NA	NA
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	NA	NA	NA	NA
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	0	NA	NA	NA	NA	NA	NA
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	NA	NA	NA	NA
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	NA	NA	NA	NA
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	NA	NA	NA	NA
Average number of radiation workers	5	NA	NA	NA	NA	NA	NA
Collective worker dose (person-rem/year)	0.71	NA	NA	NA	NA	NA	NA
Water usage (millions of liters/year)	0	NA	NA	NA	NA	NA	NA
Electricity usage (megawatt-hour/year)	1	NA	NA	NA	NA	NA	NA
Steam usage (millions of kilograms/year)	1	NA	NA	NA	NA	NA	NA
Fuel usage (thousands of liters/year)	1	NA	NA	NA	NA	NA	NA
High-level liquid waste generation (millions of liters per year)	5.7×10^{-2}	NA	NA	NA	NA	NA	NA
Equivalent DWPF canisters (per year)	1	NA	NA	NA	NA	NA	NA
Saltstone generation (cubic meters/year)	155	NA	NA	NA	NA	NA	NA
TRU waste generation (cubic meters/year)	0	NA	NA	NA	NA	NA	NA
Hazardous/mixed waste generation (cubic meters/year)	2	NA	NA	NA	NA	NA	NA
Low-level waste generation (cubic meters/year)	1,453	NA	NA	NA	NA	NA	NA

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-51. Data for impact analysis of processing and storage for vitrification (DWPF) of other aluminum-clad targets.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage[c]	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage[d]
Atmospheric MEI dose (rem/year)	1.5×10^{-8}	NA	6.6×10^{-4}	-	-	NA	-
Liquid MEI dose (rem/year)	0	NA	9.8×10^{-6}	-	-	NA	-
Total MEI dose (rem/year)	1.5×10^{-8}	NA	6.7×10^{-4}	-	-	NA	-
Atmospheric population dose (person-rem/year)[e]	9.8×10^{-4}	NA	27	-	-	NA	-
Liquid population dose (person-rem/year)[e]	0	NA	3.3×10^{-2}	-	-	NA	-
Total population dose (person-rem/year)	9.8×10^{-4}	NA	27	-	-	NA	-
Offsite CO concentration ($\mu g/m^3$) - 1-hour average[f]	0	NA	10.6	-	-	NA	-
Offsite CO concentration ($\mu g/m^3$) - 8-hour average[f]	0	NA	2.54	-	-	NA	-
Offsite NO_x concentration ($\mu g/m^3$) - annual average[f]	0	NA	8.25×10^{-2}	-	-	NA	-
Offsite SO_2 concentration ($\mu g/m^3$) - 3-hour average[f]	0	NA	6.21×10^{-3}	-	-	NA	-
Offsite SO_2 concentration ($\mu g/m^3$) - 24-hour average[f]	0	NA	1.40×10^{-3}	-	-	NA	-
Offsite SO_2 concentration ($\mu g/m^3$) - annual average[f]	0	NA	8.77×10^{-5}	-	-	NA	-
Offsite gaseous fluorides ($\mu g/m^3$) 12-hour average[f]	0	NA	7.48×10^{-6}	-	-	NA	-
Offsite gaseous fluorides ($\mu g/m^3$) 24-hour average[f]	0	NA	2.98×10^{-6}	-	-	NA	-
Offsite gaseous fluorides ($\mu g/m^3$) 1-week average[f]	0	NA	1.58×10^{-6}	-	-	NA	-
Offsite gaseous fluorides ($\mu g/m^3$) 1-month average[f]	0	NA	4.46×10^{-7}	-	-	NA	-
Offsite HNO_3 concentration ($\mu g/m^3$) - 24-hour average[f]	0	NA	2.21×10^{-2}	-	-	NA	-
Offsite HNO_3 concentration ($\mu g/m^3$) - annual average[f]	0	NA	1.39×10^{-3}	-	-	NA	-
Onsite CO concentration (mg/m^3) - 8-hour average[f]	0	NA	1.70×10^{-2}	-	-	NA	-
Onsite NO_x concentration (mg/m^3) - 1-hour average[f]	0	NA	4.36×10^{-2}	-	-	NA	-
Onsite SO_2 concentration (mg/m^3) - 8-hour average[f]	0	NA	2.46×10^{-5}	-	-	NA	-
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[f]	0	NA	3.90×10^{-4}	-	-	NA	-
Onsite CO_2 concentration (mg/m^3) - 8-hour average[f]	0	NA	1.22×10^{-5}	-	-	NA	-
Average number of radiation workers	2	NA	362	-	-	NA	-
Collective worker dose (person-rem/year)	0.21	NA	16	-	-	NA	-
Water usage (millions of liters/year)	0	NA	309	-	-	NA	-
Electricity usage (megawatt-hour/year)	1	NA	23,597	-	-	NA	-
Steam usage (millions of kilograms/year)	0	NA	141	-	-	NA	-
Fuel usage (thousands of liters/year)	0	NA	888	-	-	NA	-
High-level liquid waste generation (millions of liters per year)	1.4×10^{-2}	NA	2.2	-	-	NA	-
Equivalent DWPF canisters (per year)	0	NA	19	-	-	NA	-
Saltstone generation (cubic meters/year)	39	NA	6,086	-	-	NA	-
TRU waste generation (cubic meters/year)	0	NA	0	-	-	NA	-
Hazardous/mixed waste generation (cubic meters/year)	1	NA	8	-	-	NA	-
Low-level waste generation (cubic meters/year)	421	NA	5,693	-	-	NA	-

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Atmospheric releases from the waste tank farms in 1993 contributed less than 1×10^{-8} rem to the maximally exposed individual and less than 50 person-rem to the worker collective dose. At this time, DOE does not expect any discernible increase to these or other environmental factors as a result of this phase in the stabilization alternative.

d. DOE does not expect any discernible incremental impacts as a result of this phase in the stabilization alternative.

e. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

f. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-52. Data for impact analysis of improving storage of other aluminum-clad targets.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	1.5×10^{-8}	NA	NA	1.5×10^{-8}	NA	2.8×10^{-10}	0
Liquid MEI dose (rem/year)	0	NA	NA	0	NA	4.4×10^{-7}	0
Total MEI dose (rem/year)	1.5×10^{-8}	NA	NA	1.5×10^{-8}	NA	4.4×10^{-7}	0
Atmospheric population dose (person-rem/year)[c]	9.8×10^{-4}	NA	NA	9.8×10^{-4}	NA	1.3×10^{-5}	0
Liquid population dose (person-rem/year)[c]	0	NA	NA	0	NA	2.6×10^{-3}	0
Total population dose (person-rem/year)	9.8×10^{-4}	NA	NA	9.8×10^{-4}	NA	2.6×10^{-3}	0
Offsite CO concentration (μg/m^3) - 1-hour average[d]	0	NA	NA	0	NA	0	0
Offsite CO concentration (μg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Offsite NO_x concentration (μg/m^3) - annual average[d]	0	NA	NA	0	NA	0	0
Offsite SO_2 concentration (μg/m^3) - 3-hour average[d]	0	NA	NA	0	NA	0	0
Offsite SO_2 concentration (μg/m^3) - 24-hour average[d]	0	NA	NA	0	NA	0	0
Offsite SO_2 concentration (μg/m^3) - annual average[d]	0	NA	NA	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 12-hour average[d]	0	NA	NA	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 24-hour average[d]	0	NA	NA	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 1-week average[d]	0	NA	NA	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 1-month average[d]	0	NA	NA	0	NA	0	0
Offsite HNO_3 concentration (μg/m^3) - 24-hour average[d]	0	NA	NA	0	NA	0	0
Offsite HNO_3 concentration (μg/m^3) - annual average[d]	0	NA	NA	0	NA	0	0
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	0	NA	NA	0	NA	0	0
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Average number of radiation workers	2	NA	NA	2	NA	150	3
Collective worker dose (person-rem/year)	0.21	NA	NA	0.21	NA	6	0.13
Water usage (millions of liters/year)	0	NA	NA	0	NA	35	1
Electricity usage (megawatt-hour/year)	1	NA	NA	1	NA	1,430	0
Steam usage (millions of kilograms/year)	0	NA	NA	0	NA	11	0
Fuel usage (thousands of liters/year)	0	NA	NA	0	NA	8	0
High-level liquid waste generation (millions of liters per year)	1.4×10^{-2}	NA	NA	1.4×10^{-2}	NA	4.1×10^{-2}	0
Equivalent DWPF canisters (per year)	0	NA	NA	0	NA	0	0
Saltstone generation (cubic meters/year)	39	NA	NA	39	NA	11	0
TRU waste generation (cubic meters/year)	0	NA	NA	0	NA	0	0
Hazardous/mixed waste generation (cubic meters/year)	1	NA	NA	1	NA	0	0
Low-level waste generation (cubic meters/year)	421	NA	NA	421	NA	128	18

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

Table D-53. Data for impact analysis of improving storage (accelerated schedule) of other aluminum-clad targets.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	1.5×10^{-8}	NA	NA	1.5×10^{-8}	NA	2.8×10^{-10}	0
Liquid MEI dose (rem/year)	0	NA	NA	0	NA	4.4×10^{-7}	0
Total MEI dose (rem/year)	1.5×10^{-8}	NA	NA	1.5×10^{-8}	NA	4.4×10^{-7}	0
Atmospheric population dose (person-rem/year)[c]	9.8×10^{-4}	NA	NA	9.8×10^{-4}	NA	1.3×10^{-5}	0
Liquid population dose (person-rem/year)[c]	0	NA	NA	0	NA	2.6×10^{-3}	0
Total population dose (person-rem/year)	9.8×10^{-4}	NA	NA	9.8×10^{-4}	NA	2.6×10^{-3}	0
Offsite CO concentration (μg/m^3) - 1-hour average[d]	0	NA	NA	0	NA	0	0
Offsite CO concentration (μg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Offsite NO_x concentration (μg/m^3) - annual average[d]	0	NA	NA	0	NA	0	0
Offsite SO_2 concentration (μg/m^3) - 3-hour average[d]	0	NA	NA	0	NA	0	0
Offsite SO_2 concentration (μg/m^3) - 24-hour average[d]	0	NA	NA	0	NA	0	0
Offsite SO_2 concentration (μg/m^3) - annual average[d]	0	NA	NA	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 12-hour average[d]	0	NA	NA	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 24-hour average[d]	0	NA	NA	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 1-week average[d]	0	NA	NA	0	NA	0	0
Offsite gaseous fluorides (μg/m^3) 1-month average[d]	0	NA	NA	0	NA	0	0
Offsite HNO_3 concentration (μg/m^3) - 24-hour average[d]	0	NA	NA	0	NA	0	0
Offsite HNO_3 concentration (μg/m^3) - annual average[d]	0	NA	NA	0	NA	0	0
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	0	NA	NA	0	NA	0	0
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Average number of radiation workers	2	NA	NA	2	NA	150	3
Collective worker dose (person-rem/year)	0.21	NA	NA	0.21	NA	6	0.13
Water usage (millions of liters/year)	0	NA	NA	0	NA	35	1
Electricity usage (megawatt-hour/year)	1	NA	NA	1	NA	1,430	0
Steam usage (millions of kilograms/year)	0	NA	NA	0	NA	11	0
Fuel usage (thousands of liters/year)	0	NA	NA	0	NA	8	0
High-level liquid waste generation (millions of liters per year)	1.4×10^{-2}	NA	NA	1.4×10^{-2}	NA	4.1×10^{-2}	0
Equivalent DWPF canisters (per year)	0	NA	NA	0	NA	0	0
Saltstone generation (cubic meters/year)	39	NA	NA	39	NA	11	0
TRU waste generation (cubic meters/year)	0	NA	NA	0	NA	0	0
Hazardous/mixed waste generation (cubic meters/year)	1	NA	NA	1	NA	0	0
Low-level waste generation (cubic meters/year)	421	NA	NA	421	NA	128	18

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-54. Data for impact analysis of continuing storage (No Action) of other aluminum-clad targets.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	1.5×10^{-8}	NA	NA	NA	NA	NA	NA
Liquid MEI dose (rem/year)	0	NA	NA	NA	NA	NA	NA
Total MEI dose (rem/year)	1.5×10^{-8}	NA	NA	NA	NA	NA	NA
Atmospheric population dose (person-rem/year)[c]	9.8×10^{-4}	NA	NA	NA	NA	NA	NA
Liquid population dose (person-rem/year)[c]	0	NA	NA	NA	NA	NA	NA
Total population dose (person-rem/year)	9.8×10^{-4}	NA	NA	NA	NA	NA	NA
Offsite CO concentration ($\mu g/m^3$) - 1-hour averaged[d]	0	NA	NA	NA	NA	NA	NA
Offsite CO concentration ($\mu g/m^3$) - 8-hour averaged[d]	0	NA	NA	NA	NA	NA	NA
Offsite NO_x concentration ($\mu g/m^3$) - annual averaged[d]	0	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration ($\mu g/m^3$) - 3-hour averaged[d]	0	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration ($\mu g/m^3$) - 24-hour averaged[d]	0	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration ($\mu g/m^3$) - annual averaged[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 12-hour averaged[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 24-hour averaged[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 1-week averaged[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides ($\mu g/m^3$) 1-month averaged[d]	0	NA	NA	NA	NA	NA	NA
Offsite HNO_3 concentration ($\mu g/m^3$) - 24-hour averaged[d]	0	NA	NA	NA	NA	NA	NA
Offsite HNO_3 concentration ($\mu g/m^3$) - annual averaged[d]	0	NA	NA	NA	NA	NA	NA
Onsite CO concentration (mg/m^3) - 8-hour averaged[d]	0	NA	NA	NA	NA	NA	NA
Onsite NO_x concentration (mg/m^3) - 1-hour averaged[d]	0	NA	NA	NA	NA	NA	NA
Onsite SO_2 concentration (mg/m^3) - 8-hour averaged[d]	0	NA	NA	NA	NA	NA	NA
Onsite HNO_3 concentration (mg/m^3) - 8-hour averaged[d]	0	NA	NA	NA	NA	NA	NA
Onsite CO_2 concentration (mg/m^3) - 8-hour averaged[d]	0	NA	NA	NA	NA	NA	NA
Average number of radiation workers	2	NA	NA	NA	NA	NA	NA
Collective worker dose (person-rem/year)	0.21	NA	NA	NA	NA	NA	NA
Water usage (millions of liters/year)	0	NA	NA	NA	NA	NA	NA
Electricity usage (megawatt-hour/year)	1	NA	NA	NA	NA	NA	NA
Steam usage (millions of kilograms/year)	0	NA	NA	NA	NA	NA	NA
Fuel usage (thousands of liters/year)	0	NA	NA	NA	NA	NA	NA
High-level liquid waste generation (millions of liters per year)	1.4×10^{-2}	NA	NA	NA	NA	NA	NA
Equivalent DWPF canisters (per year)	0	NA	NA	NA	NA	NA	NA
Saltstone generation (cubic meters/year)	39	NA	NA	NA	NA	NA	NA
TRU waste generation (cubic meters/year)	0	NA	NA	NA	NA	NA	NA
Hazardous/mixed waste generation (cubic meters/year)	1	NA	NA	NA	NA	NA	NA
Low-level waste generation (cubic meters/year)	421	NA	NA	NA	NA	NA	NA

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides, SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for (air or liquid) pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-55. Data for impact analysis of continuing storage (No Action) of Taiwan Research Reactor Fuel and Experimental Breeder Reactor-II slugs.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	1.4×10^{-10}	NA	NA	NA	NA	NA	NA
Liquid MEI dose (rem/year)	3.5×10^{-7}	NA	NA	NA	NA	NA	NA
Total MEI dose (rem/year)	3.5×10^{-7}	NA	NA	NA	NA	NA	NA
Atmospheric population dose (person-rem/year)[c]	5.8×10^{-6}	NA	NA	NA	NA	NA	NA
Liquid population dose (person-rem/year)[c]	1.0×10^{-3}	NA	NA	NA	NA	NA	NA
Total population dose (person-rem/year)	1.0×10^{-3}	NA	NA	NA	NA	NA	NA
Offsite CO concentration (mg/m^3) - 1-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite NO_x concentration (mg/m^3) - annual average[d]	0	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration (mg/m^3) - 3-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration (mg/m^3) - 24-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite SO_2 concentration (mg/m^3) - annual average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (mg/m^3) 12-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (mg/m^3) 24-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (mg/m^3) 1-week average[d]	0	NA	NA	NA	NA	NA	NA
Offsite gaseous fluorides (mg/m^3) 1-month average[d]	0	NA	NA	NA	NA	NA	NA
Offsite HNO_3 concentration (mg/m^3) - 24-hour average[d]	0	NA	NA	NA	NA	NA	NA
Offsite HNO_3 concentration (mg/m^3) - annual average[d]	0	NA	NA	NA	NA	NA	NA
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	NA	NA	NA	NA
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	0	NA	NA	NA	NA	NA	NA
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	NA	NA	NA	NA
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	NA	NA	NA	NA
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	NA	NA	NA	NA
Average number of radiation workers	34	NA	NA	NA	NA	NA	NA
Collective worker dose (person-rem/year)	1.4	NA	NA	NA	NA	NA	NA
Water usage (millions of liters per year)	28	NA	NA	NA	NA	NA	NA
Electricity usage (megawatt-hours/year)	1,144	NA	NA	NA	NA	NA	NA
Steam usage (millions of kilograms per year)	9	NA	NA	NA	NA	NA	NA
Fuel usage (thousands of liters per year)	6	NA	NA	NA	NA	NA	NA
High-level liquid waste generation (millions of liters per year)	0.04	NA	NA	NA	NA	NA	NA
Equivalent DWPF canisters (per year)	0	NA	NA	NA	NA	NA	NA
Saltstone generation (cubic meters/year)	11	NA	NA	NA	NA	NA	NA
Tru waste generation (cubic meters/year)	0	NA	NA	NA	NA	NA	NA
Hazardous/mixed waste generation (cubic meters/year)	0	NA	NA	NA	NA	NA	NA
Low-level waste generation (cubic meters/year)	130	NA	NA	NA	NA	NA	NA

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; CO_2 = carbon dioxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides; SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for air or liquid pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-56. Data for impact analysis of processing Taiwan Research Reactor Fuel and Experimental Breeder Reactor-II slugs to metal.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	1.4×10^{-10}	NA	1.1×10^{-5}	2.3×10^{-9}	NA	6.6×10^{-9}	4.4×10^{-9}
Liquid MEI dose (rem/year)	3.5×10^{-7}	NA	7.4×10^{-7}	2.2×10^{-10}	NA	7.3×10^{-10}	4.3×10^{-10}
Total MEI dose (rem/year)	3.5×10^{-7}	NA	1.1×10^{-5}	2.5×10^{-9}	NA	7.3×10^{-9}	7.3×10^{-9}
Atmospheric population dose (person-rem/year)[c]	5.8×10^{-6}	NA	4.8×10^{-1}	1.0×10^{-4}	NA	2.9×10^{-4}	2.0×10^{-4}
Liquid population dose (person-rem/year)[c]	1.0×10^{-3}	NA	3.3×10^{-3}	3.5×10^{-6}	NA	6.4×10^{-6}	6.8×10^{-6}
Total population dose (person-rem/year)	1.0×10^{-3}	NA	4.8×10^{-1}	1.1×10^{-4}	NA	3.0×10^{-4}	2.0×10^{-4}
Offsite CO concentration (mg/m^3) - 1-hour average[d]	0	NA	0	0	NA	0	0
Offsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	NA	0	0
Offsite NO_X concentration (mg/m^3) - annual average[d]	0	NA	0.283	0	NA	0	0
Offsite SO_2 concentration (mg/m^3) - 3-hour average[d]	0	NA	0	0	NA	0	0
Offsite SO_2 concentration (mg/m^3) - 24-hour average[d]	0	NA	0	0	NA	0	0
Offsite SO_2 concentration (mg/m^3) - annual average[d]	0	NA	0	0	NA	0	0
Offsite gaseous fluorides (mg/m^3) 12-hour average[d]	0	NA	5.24×10^{-2}	0	NA	0	0
Offsite gaseous fluorides (mg/m^3) 24-hour average[d]	0	NA	2.82×10^{-2}	0	NA	0	0
Offsite gaseous fluorides (mg/m^3) 1-week average[d]	0	NA	1.11×10^{-2}	0	NA	0	0
Offsite gaseous fluorides (mg/m^3) 1-month average[d]	0	NA	3.12×10^{-3}	0	NA	0	0
Offsite HNO_3 concentration (mg/m^3) - 24-hour average[d]	0	NA	0.777	0	NA	0	0
Offsite HNO_3 concentration (mg/m^3) - annual average[d]	0	NA	5.27×10^{-2}	0	NA	0	0
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	NA	0	0
Onsite NO_X concentration (mg/m^3) - 1-hour average[d]	0	NA	0.179	0	NA	0	0
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	NA	0	0
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	0	NA	1.22×10^{-2}	0	NA	0	0
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	NA	0	0
Average number of radiation workers	34	NA	776	47	NA	157	80
Collective worker dose (person-rem/year)	1.4	NA	160	2.0	NA	31	16
Water usage (millions of liters per year)	28	NA	1,534	278	NA	500	454
Electricity usage (megawatt-hours/year)	1,144	NA	29,150	3,683	NA	6,620	6,018
Steam usage (millions of kilograms per year)	9	NA	147	4	NA	7	6
Fuel usage (thousands of liters per year)	6	NA	893	12	NA	21	19
High-level liquid waste generation (millions of liters per year)	0.04	NA	1.9	0	NA	0	0
Equivalent DWPF canisters (per year)	0	NA	38	0	NA	0	0
Saltstone generation (cubic meters/year)	11	NA	5,090	0	NA	0	0
Tru waste generation (cubic meters/year)	0	NA	64	2	NA	1	0
Hazardous/mixed waste generation (cubic meters/year)	0	NA	0	1	NA	1	0
Low-level waste generation (cubic meters/year)	130	NA	6,160	970	NA	7	5

TC

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; CO_2 = carbon dioxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_X = nitrogen oxides; SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for air or liquid pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

Table D-57. Data for impact analysis of processing Taiwan Research Reactor Fuel and Experimental Breeder Reactor-II slugs to oxide.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	1.4×10^{-10}	NA	1.1×10^{-5}	2.3×10^{-9}	NA	6.6×10^{-9}	4.4×10^{-9}
Liquid MEI dose (rem/year)	3.5×10^{-7}	NA	7.4×10^{-7}	2.2×10^{-10}	NA	7.3×10^{-10}	4.3×10^{-10}
Total MEI dose (rem/year)	3.5×10^{-7}	NA	1.1×10^{-5}	2.5×10^{-9}	NA	7.3×10^{-9}	4.8×10^{-9}
Atmospheric population dose (person-rem/year)[c]	5.8×10^{-6}	NA	4.8×10^{-1}	1.0×10^{-4}	NA	2.9×10^{-4}	2.0×10^{-4}
Liquid population dose (person-rem/year)[c]	1.0×10^{-3}	NA	3.3×10^{-3}	3.5×10^{-6}	NA	6.4×10^{-6}	6.8×10^{-6}
Total population dose (person-rem/year)	1.0×10^{-3}	NA	4.8×10^{-1}	1.1×10^{-4}	NA	3.0×10^{-4}	2.0×10^{-4}
Offsite CO concentration (mg/m^3) - 1-hour average[d]	0	NA	0	0	NA	0	0
Offsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	NA	0	0
Offsite NO_x concentration (mg/m^3) - annual average[d]	0	NA	0.283	0	NA	0	0
Offsite SO_2 concentration (mg/m^3) - 3-hour average[d]	0	NA	0	0	NA	0	0
Offsite SO_2 concentration (mg/m^3) - 24-hour average[d]	0	NA	0	0	NA	0	0
Offsite SO_2 concentration (mg/m^3) - annual average[d]	0	NA	0	0	NA	0	0
Offsite gaseous fluorides (mg/m^3) 12-hour average[d]	0	NA	5.24×10^{-2}	0	NA	0	0
Offsite gaseous fluorides (mg/m^3) 24-hour average[d]	0	NA	2.82×10^{-2}	0	NA	0	0
Offsite gaseous fluorides (mg/m^3) 1-week average[d]	0	NA	1.11×10^{-2}	0	NA	0	0
Offsite gaseous fluorides (mg/m^3) 1-month average[d]	0	NA	3.12×10^{-3}	0	NA	0	0
Offsite HNO_3 concentration (mg/m^3) - 24-hour average[d]	0	NA	0.777	0	NA	0	0
Offsite HNO_3 concentration (mg/m^3) - annual average[d]	0	NA	5.27×10^{-2}	0	NA	0	0
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	NA	0	0
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	0	NA	0.179	0	NA	0	0
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	NA	0	0
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	0	NA	1.22×10^{-2}	0	NA	0	0
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	NA	0	0
Average number of radiation workers	34	NA	776	47	NA	157	80
Collective worker dose (person-rem/year)	1.4	NA	160	2.0	NA	31	16
Water usage (millions of liters per year)	28	NA	1,534	278	NA	500	454
Electricity usage (megawatt-hours/year)	1,144	NA	29,150	3,683	NA	6,620	6,018
Steam usage (millions of kilograms per year)	9	NA	147	4	NA	7	6
Fuel usage (thousands of liters per year)	6	NA	893	12	NA	21	19
High-level liquid waste generation (millions of liters per year)	0.04	NA	1.9	0	NA	0	0
Equivalent DWPF canisters (per year)	0	NA	38	0	NA	0	0
Saltstone generation (cubic meters/year)	11	NA	5,090	0	NA	0	0
Tru waste generation (cubic meters/year)	0	NA	64	2	NA	1	0
Hazardous/mixed waste generation (cubic meters/year)	0	NA	0	1	NA	1	0
Low-level waste generation (cubic meters/year)	130	NA	6,160	970	NA	7	5

TC

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; CO_2 = carbon dioxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides; SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for air or liquid pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

Table D-58. Data for impact analysis of processing and storage for vitrification (DWPF) of Taiwan Research Reactor Fuel and Experimental Breeder Reactor-II slugs.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage[c]	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage[d]
Atmospheric MEI dose (rem/year)	1.4×10^{-10}	NA	8.5×10^{-6}	-	1.6×10^{-4}	NA	-
Liquid MEI dose (rem/year)	3.5×10^{-7}	NA	6.8×10^{-7}	-	2.4×10^{-5}	NA	-
Total MEI dose (rem/year)	3.5×10^{-7}	NA	9.2×10^{-6}	-	1.8×10^{-4}	NA	-
Atmospheric population dose (person-rem/year)[c]	5.8×10^{-6}	NA	3.8×10^{-1}	-	6.9	NA	-
Liquid population dose (person-rem/year)[c]	1.0×10^{-3}	NA	2.9×10^{-3}	-	1.4×10^{-1}	NA	-
Total population dose (person-rem/year)	1.0×10^{-3}	NA	3.8×10^{-1}	-	7.0	NA	-
Offsite CO concentration (mg/m^3) - 1-hour average[d]	0	NA	0	-	0	NA	-
Offsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	0	-	0	NA	-
Offsite NO_x concentration (mg/m^3) - annual average[d]	0	NA	0.227	-	0	NA	-
Offsite SO_2 concentration (mg/m^3) - 3-hour average[d]	0	NA	0	-	0	NA	-
Offsite SO_2 concentration (mg/m^3) - 24-hour average[d]	0	NA	0	-	0	NA	-
Offsite SO_2 concentration (mg/m^3) - annual average[d]	0	NA	0	-	0	NA	-
Offsite gaseous fluorides (mg/m^3) 12-hour average[d]	0	NA	4.19×10^{-2}	-	0	NA	-
Offsite gaseous fluorides (mg/m^3) 24-hour average[d]	0	NA	2.26×10^{-2}	-	0	NA	-
Offsite gaseous fluorides (mg/m^3) 1-week average[d]	0	NA	8.87×10^{-3}	-	0	NA	-
Offsite gaseous fluorides (mg/m^3) 1-month average[d]	0	NA	2.50×10^{-3}	-	0	NA	-
Offsite HNO_3 concentration (mg/m^3) - 24-hour average[d]	0	NA	0.622	-	0	NA	-
Offsite HNO_3 concentration (mg/m^3) - annual average[d]	0	NA	4.22×10^{-2}	-	0	NA	-
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	0	-	0	NA	-
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	0	NA	0.143	-	0	NA	-
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	0	-	0	NA	-
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	0	NA	9.77×10^{-3}	-	0	NA	-
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	0	-	0	NA	-
Average number of radiation workers	34	NA	395	-	1,500	NA	-
Collective worker dose (person-rem/year)	1.4	NA	54	-	60	NA	-
Water usage (millions of liters per year)	28	NA	1,234	-	1,476	NA	-
Electricity usage (megawatt-hours/year)	1,144	NA	23,606	-	50,371	NA	-
Steam usage (millions of kilograms per year)	9	NA	120	-	167	NA	-
Fuel usage (thousands of liters per year)	6	NA	716	-	0	NA	-
High-level liquid waste generation (millions of liters per year)	0.04	NA	1.5	-	0	NA	-
Equivalent DWPF canisters (per year)	0	NA	29	-	405	NA	-
Saltstone generation (cubic meters/year)	11	NA	3,900	-	0	NA	-
Tru waste generation (cubic meters/year)	0	NA	0	-	1,140	NA	-
Hazardous/mixed waste generation (cubic meters/year)	0	NA	0	-	3,060	NA	-
Low-level waste generation (cubic meters/year)	130	NA	1,290	-	2,730	NA	-

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; CO2 = carbon dioxide; DWPF = Defense Waste Processing Facility; HNO3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NOx = nitrogen oxides; SO2 = sulfur dioxide; TRU = transuranic.

c. Atmospheric releases from the waste tank farms in 1993 contributed less than 1×10^{-8} rem to the maximally exposed individual and less than 50 person-rem to the worker collective dose. At this time, DOE does not expect any discernible increase to these or other environmental factors as a result of this phase in the stabilization alternative.

d. DOE does not expect any discernible incremental impacts as a result of this phase in the stabilization alternative.

e. Data from WSRC (1995a). MEI dose value for air or liquid pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

f. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-59. Data for impact analysis of vitrification (F-Canyon) of Taiwan Research Reactor Fuel and Experimental Breeder Reactor-II slugs.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	1.4×10^{-10}	NA	1.1×10^{-5}	2.3×10^{-9}	1.3×10^{-5}	NA	4.4×10^{-9}
Liquid MEI dose (rem/year)	3.5×10^{-7}	NA	7.4×10^{-7}	2.2×10^{-10}	3.8×10^{-7}	NA	4.3×10^{-10}
Total MEI dose (rem/year)	3.5×10^{-7}	NA	1.1×10^{-5}	2.5×10^{-9}	1.3×10^{-5}	NA	4.8×10^{-9}
Atmospheric population dose (person-rem/year)[c]	5.8×10^{-6}	NA	4.8×10^{-1}	1.0×10^{-4}	5.7×10^{-1}	NA	2.0×10^{-4}
Liquid population dose (person-rem/year)[c]	1.0×10^{-3}	NA	3.3×10^{-3}	3.5×10^{-6}	2.2×10^{-3}	NA	6.8×10^{-6}
Total population dose (person-rem/year)	1.0×10^{-3}	NA	4.8×10^{-1}	1.1×10^{-4}	5.7×10^{-1}	NA	2.0×10^{-4}
Offsite CO concentration (mg/m^3) - 1-hour average[d]	0	NA	0	0	0	NA	0
Offsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	0	NA	0
Offsite NO_x concentration (mg/m^3) - annual average[d]	0	NA	0.283	0	0.340	NA	0
Offsite SO_2 concentration (mg/m^3) - 3-hour average[d]	0	NA	0	0	0	NA	0
Offsite SO_2 concentration (mg/m^3) - 24-hour average[d]	0	NA	0	0	0	NA	0
Offsite SO_2 concentration (mg/m^3) - annual average[d]	0	NA	0	0	0	NA	0
Offsite gaseous fluorides (mg/m^3) 12-hour average[d]	0	NA	5.24×10^{-2}	0	6.29×10^{-2}	NA	0
Offsite gaseous fluorides (mg/m^3) 24-hour average[d]	0	NA	2.82×10^{-2}	0	3.39×10^{-2}	NA	0
Offsite gaseous fluorides (mg/m^3) 1-week average[d]	0	NA	1.11×10^{-2}	0	1.33×10^{-2}	NA	0
Offsite gaseous fluorides (mg/m^3) 1-month average[d]	0	NA	3.12×10^{-3}	0	3.75×10^{-3}	NA	0
Offsite HNO_3 concentration (mg/m^3) - 24-hour average[d]	0	NA	0.777	0	0.933	NA	0
Offsite HNO_3 concentration (mg/m^3) - annual average[d]	0	NA	5.27×10^{-2}	0	6.33×10^{-2}	NA	0
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	0	NA	0
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	0	NA	0.179	0	2.15×10^{-2}	NA	0
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	0	NA	0
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	0	NA	1.22×10^{-2}	0	1.47×10^{-2}	NA	0
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	0	0	0	NA	0
Average number of radiation workers	34	NA	776	47	790	NA	80
Collective worker dose (person-rem/year)	1.4	NA	160	2.0	170	NA	16
Water usage (millions of liters per year)	28	NA	1,534	278	1,799	NA	454
Electricity usage (megawatt-hours/year)	1,144	NA	29,150	3,683	33,264	NA	6,018
Steam usage (millions of kilograms per year)	9	NA	147	4	163	NA	6
Fuel usage (thousands of liters per year)	6	NA	893	12	1,063	NA	19
High-level liquid waste generation (millions of liters per year)	0.04	NA	1.9	0	1.9	NA	0
Equivalent DWPF canisters (per year)	0	NA	38	0	38	NA	0
Saltstone generation (cubic meters/year)	11	NA	5,090	0	5,080	NA	0
Tru waste generation (cubic meters/year)	0	NA	64	2	64	NA	0
Hazardous/mixed waste generation (cubic meters/year)	0	NA	0	1	0	NA	0
Low-level waste generation (cubic meters/year)	130	NA	6,160	970	6,150	NA	5

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; CO_2 = carbon dioxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides; SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for air or liquid pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-60. Data for impact analysis of improving storage of Taiwan Research Reactor Fuel and Experimental Breeder Reactor-II slugs.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	1.4×10^{-10}	NA	NA	1.4×10^{-10}	NA	2.8×10^{-10}	0
Liquid MEI dose (rem/year)	3.5×10^{-7}	NA	NA	3.5×10^{-7}	NA	4.4×10^{-7}	0
Total MEI dose (rem/year)	3.5×10^{-7}	NA	NA	3.5×10^{-7}	NA	4.4×10^{-7}	0
Atmospheric population dose (person-rem/year)[c]	5.8×10^{-6}	NA	NA	5.8×10^{-6}	NA	1.3×10^{-5}	0
Liquid population dose (person-rem/year)[c]	1.0×10^{-3}	NA	NA	1.0×10^{-3}	NA	2.6×10^{-3}	0
Total population dose (person-rem/year)	1.0×10^{-3}	NA	NA	1.0×10^{-3}	NA	2.6×10^{-3}	0
Offsite CO concentration (mg/m^3) - 1-hour average[d]	0	NA	NA	0	NA	0	0
Offsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Offsite NO_x concentration (mg/m^3) - annual average[d]	0	NA	NA	0	NA	0	0
Offsite SO_2 concentration (mg/m^3) - 3-hour average[d]	0	NA	NA	0	NA	0	0
Offsite SO_2 concentration (mg/m^3) - 24-hour average[d]	0	NA	NA	0	NA	0	0
Offsite SO_2 concentration (mg/m^3) - annual average[d]	0	NA	NA	0	NA	0	0
Offsite gaseous fluorides (mg/m^3) 12-hour average[d]	0	NA	NA	0	NA	0	0
Offsite gaseous fluorides (mg/m^3) 24-hour average[d]	0	NA	NA	0	NA	0	0
Offsite gaseous fluorides (mg/m^3) 1-week average[d]	0	NA	NA	0	NA	0	0
Offsite gaseous fluorides (mg/m^3) 1-month average[d]	0	NA	NA	0	NA	0	0
Offsite HNO_3 concentration (mg/m^3) - 24-hour average[d]	0	NA	NA	0	NA	0	0
Offsite HNO_3 concentration (mg/m^3) - annual average[d]	0	NA	NA	0	NA	0	0
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	0	NA	NA	0	NA	0	0
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Average number of radiation workers	34	NA	NA	34	NA	150	3.0
Collective worker dose (person-rem/year)	1.4	NA	NA	1.4	NA	6.0	0.13
Water usage (millions of liters per year)	28	NA	NA	28	NA	35	1
Electricity usage (megawatt-hours/year)	1,144	NA	NA	1,144	NA	1,430	0
Steam usage (millions of kilograms per year)	9	NA	NA	9	NA	11	0
Fuel usage (thousands of liters per year)	6	NA	NA	6	NA	6	0
High-level liquid waste generation (millions of liters per year)	0.04	NA	NA	0.04	NA	0.04	0
Equivalent DWPF canisters (per year)	0	NA	NA	0	NA	0	0
Saltstone generation (cubic meters/year)	11	NA	NA	11	NA	11	0
Tru waste generation (cubic meters/year)	0	NA	NA	0	NA	0	0
Hazardous/mixed waste generation (cubic meters/year)	0	NA	NA	0	NA	0	0
Low-level waste generation (cubic meters/year)	130	NA	NA	130	NA	130	0

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; CO_2 = carbon dioxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides; SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for air or liquid pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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Table D-61. Data for impact analysis of improving storage (accelerated schedule) of Taiwan Research Reactor Fuel and Experimental Breeder Reactor-II slugs.[a,b]

Factor	Existing storage	Characterization	Conversion	Interim storage	Additional conversion (if required)	Packaging/ repackaging	Post-stabilization storage
Atmospheric MEI dose (rem/year)	1.4×10^{-10}	NA	NA	1.4×10^{-10}	NA	2.8×10^{-10}	0
Liquid MEI dose (rem/year)	3.5×10^{-7}	NA	NA	3.5×10^{-7}	NA	4.4×10^{-7}	0
Total MEI dose (rem/year)	3.5×10^{-7}	NA	NA	3.5×10^{-7}	NA	4.4×10^{-7}	0
Atmospheric population dose (person-rem/year)[c]	5.8×10^{-6}	NA	NA	5.8×10^{-6}	NA	1.3×10^{-5}	0
Liquid population dose (person-rem/year)[c]	1.0×10^{-3}	NA	NA	1.0×10^{-3}	NA	2.6×10^{-3}	0
Total population dose (person-rem/year)	1.0×10^{-3}	NA	NA	1.0×10^{-3}	NA	2.6×10^{-3}	0
Offsite CO concentration (mg/m^3) - 1-hour average[d]	0	NA	NA	0	NA	0	0
Offsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Offsite NO_x concentration (mg/m^3) - annual average[d]	0	NA	NA	0	NA	0	0
Offsite SO_2 concentration (mg/m^3) - 3-hour average[d]	0	NA	NA	0	NA	0	0
Offsite SO_2 concentration (mg/m^3) - 24-hour average[d]	0	NA	NA	0	NA	0	0
Offsite SO_2 concentration (mg/m^3) - annual average[d]	0	NA	NA	0	NA	0	0
Offsite gaseous fluorides (mg/m^3) 12-hour average[d]	0	NA	NA	0	NA	0	0
Offsite gaseous fluorides (mg/m^3) 24-hour average[d]	0	NA	NA	0	NA	0	0
Offsite gaseous fluorides (mg/m^3) 1-week average[d]	0	NA	NA	0	NA	0	0
Offsite gaseous fluorides (mg/m^3) 1-month average[d]	0	NA	NA	0	NA	0	0
Offsite HNO_3 concentration (mg/m^3) - 24-hour average[d]	0	NA	NA	0	NA	0	0
Offsite HNO_3 concentration (mg/m^3) - annual average[d]	0	NA	NA	0	NA	0	0
Onsite CO concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Onsite NO_x concentration (mg/m^3) - 1-hour average[d]	0	NA	NA	0	NA	0	0
Onsite SO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Onsite HNO_3 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Onsite CO_2 concentration (mg/m^3) - 8-hour average[d]	0	NA	NA	0	NA	0	0
Average number of radiation workers	34	NA	NA	34	NA	150	17
Collective worker dose (person-rem/year)	1.4	NA	NA	1.4	NA	6.0	0.66
Water usage (millions of liters per year)	28	NA	NA	28	NA	35	1
Electricity usage (megawatt-hours/year)	1,144	NA	NA	1,144	NA	1,430	0
Steam usage (millions of kilograms per year)	9	NA	NA	9	NA	11	0
Fuel usage (thousands of liters per year)	6	NA	NA	6	NA	6	0
High-level liquid waste generation (millions of liters per year)	0.04	NA	NA	0.04	NA	0.04	0
Equivalent DWPF canisters (per year)	0	NA	NA	0	NA	0	0
Saltstone generation (cubic meters/year)	11	NA	NA	11	NA	11	0
Tru waste generation (cubic meters/year)	0	NA	NA	0	NA	0	0
Hazardous/mixed waste generation (cubic meters/year)	0	NA	NA	0	NA	0	0
Low-level waste generation (cubic meters/year)	130	NA	NA	130	NA	130	92

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a. Source: WSRC (1995a).

b. Abbreviations: CO = carbon monoxide; CO_2 = carbon dioxide; DWPF = Defense Waste Processing Facility; HNO_3 = nitric acid; MEI = maximally exposed individual; NA = not applicable; NO_x = nitrogen oxides; SO_2 = sulfur dioxide; TRU = transuranic.

c. Data from WSRC (1995a). MEI dose value for air or liquid pathway was used to determine source emission rate (number of curies released per year). Conversion factors for emissions (per unit curie) from Simpkins (1994a).

d. Information on release rates of nonradiological pollutants from WSRC (1995a). Modeling data and conversion factors from Hess (1995).

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APPENDIX E

ACCIDENTS

TABLE OF CONTENTS

Section Page

E.1 General Accident Information E-1

E.2 Accident Analysis Method E-3

- E.2.1 Affected Facilities E-6
- E.2.2 Radiological Accident Analysis Method E-6
- E.2.3 Hazardous Material Accident Analysis Methodology E-9

E.3 Postulated Accidents Involving Radioactive Materials E-12

- E.3.1 Impacts to Uninvolved Workers and Members of the Public E-12
- E.3.2 Impacts to Facility Workers from Postulated Facility Accidents E-55
  - E.3.2.1 F-Canyon and H-Canyon E-55
  - E.3.2.2 FB-Line Facility E-56
  - E.3.2.3 FA-Line E-56
  - E.3.2.4 235-F Storage Vaults E-56
  - E.3.2.5 HB-Line Facility E-57
  - E.3.2.6 Uranium Solidification Facility E-57
  - E.3.2.7 H-Area Receiving Basin for Offsite Fuel E-57
  - E.3.2.8 Reactor Disassembly Basins E-58
  - E.3.2.9 Other Facilities E-58
- E.3.3 Stable Materials E-59
  - E.3.3.1 Postulated Radiological Accidents for the M-Area Reactor Materials Facilities E-59
  - E.3.3.2 Postulated Radiological Accidents for Savannah River Technology Center E-60
  - E.3.3.3 Postulated Radiological Accidents for the TNX Research Facility E-60

E.4 Postulated Accidents Involving Extremely Hazardous Substances E-61

- E.4.1 Postulated Chemical Accidents for F-Area Facilities E-61
- E.4.2 Postulated Chemical Accidents for H-Area Facilities E-64
- E.4.3 Postulated Chemical Accidents for K-, L-, and P-Reactor Basins E-66
- E.4.4 Postulated Chemical Accidents for M-Area Facilities E-67
- E.4.5 Savannah River Technology Center E-67
- E.4.6 Postulated Chemical Accidents for the TNX Area E-68
- E.4.7 Postulated Chemical Accidents for the F- and H-Area Tank Farm E-68
- E.4.8 Postulated Chemical Accidents for the Defense Waste Processing Facility and In-Tank Precipitation E-70

E.5 Environmental Justice E-71

E.6 Accident Mitigation E-72

TABLE OF CONTENTS (continued)

Section Page

E.7 Nuclear Criticality Safety Program E-74

E.8 Accident Descriptions E-76

LIST OF TABLES

Table Page

E-1 Accident frequency categories E-2
E-2 Hierarchy of established limits and guidelines used to determine impacts from postulated hazardous material accidents E-11
E-3 Applicable facilities for each alternative E-14
E-4 Plutonium-242 .... E-17
E-5 Americium and curium E-22
E-6 Neptunium E-25
E-7 H-Canyon plutonium-239 solutions E-29
E-8 H-Canyon enriched uranium solutions E-33
E-9 Plutonium and uranium stored in vaults E-37
E-10 Mark-31 targets E-43
E-11 Mark-16 and -22 fuel E-48
E-12 Taiwan Research Reactor and Experimental Breeder Reactor-II fuel E-51
E-13 Inventories of Extremely Hazardous Substances in F-Area E-62
E-14 Impacts from potential non-seismic-initiated releases of extremely hazardous substances in F-Area E-63
E-15 Impacts from potential releases of extremely hazardous substances in F-Area resulting from a severe earthquake E-63
E-16 Inventories of Extremely Hazardous Substances in H-Area E-65
E-17 Impacts from potential non-seismic-initiated releases of Extremely Hazardous Substances in H-Area E-65
E-18 Impacts from potential releases of extremely hazardous substances in H-Area resulting from a severe earthquake E-66
E-19 Inventories of Extremely Hazardous Substances in M-Area E-67
E-20 Inventories of Extremely Hazardous Substances in Savannah River Technology Center E-68
E-21 Hazardous chemical inventory (designated as extremely hazardous substances) for the waste tank farms E-69
E-22 Summary of hazardous chemical assessment accident analysis results for the waste tank farms E-69
E-23 Summary of the vitrification facility chemical hazard comparisons E-70
E-24 Summary of in-tank precipitation accident analysis results E-71

LIST OF FIGURES

Figure Page

E-1 Example of a fault tree E-4
E-2 Methodology used to determine bounding risk accidents for the various nuclear facilities E-13

APPENDIX E. ACCIDENTS

This appendix summarizes accidents that could involve nuclear material management. It provides consequences (e.g., resulting doses) from potential releases of specific nuclear materials for each alternative discussed in this EIS.

In preparing this environmental impact statement, DOE reviewed safety analysis reports and supporting accident analyses for facilities that the alternatives described in Chapter 2 could involve. There are no accident analyses for alternatives that would involve new facilities or extensive modifications to existing facilities. In such cases, DOE used accident analyses for existing facilities at SRS that perform similar operations or that process and handle forms of nuclear material that are more hazardous than those being considered in this EIS. DOE believes that the types of accidents evaluated for such existing facilities would be comparable to those for new or modified facilities.

E.1 General Accident Information

An "accident," as discussed in this appendix, is an unplanned release of radioactive or hazardous materials resulting from "initiating" events and the additional failures resulting from the initiating event. In this case, an accident is an inadvertent release of radioactive or hazardous materials from their containers or confinement to the environment.[1] Initiating events are typically defined in three broad categories:

- External initiators originate outside the facility and potentially affect the ability of the facility to maintain confinement of its materials. Examples of external initiators include aircraft crashes, nearby explosions, and hazardous material releases from nearby facilities that could affect the ability of personnel to manage the facility and its materials properly.
- Internal initiators originate within a facility and are usually the result of facility operation. Examples of internal initiators include equipment failures and human errors.
- Natural phenomena initiators are natural occurrences such as weather-related (e.g., floods and tornadoes) and seismic events (i.e., earthquakes).

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[1] For this appendix, "environment" includes areas within a facility occupied by workers and outward from the facility where the release occurs.

The likelihood of an accident occurring and its consequences usually depend on the type of initiator(s) causing the accident, the frequency at which that initiator occurs, and the frequency of conditions that will lead to a release caused by the initiating event. Accidents can be grouped into four categories -- anticipated accidents, unlikely accidents, extremely unlikely accidents, and not reasonably foreseeable accidents -- based on their estimated frequency or likelihood of occurrence. Table E-1 lists these accident categories and their corresponding frequency ranges.

Table E-1. Accident frequency categories.[a]

Frequency category	Frequency range (incidents per year)	Description
1. Anticipated accidents	Less than once in 10 years but greater than once in 100 years	Accidents that might occur several times during the lifetime of the facility.
2. Unlikely accidents	Less than once in 100 years but greater than once in 10,000 years	Accidents that are not likely to occur during the lifetime of the facility; natural phenomena of this probability class include Uniform Building Code-level earthquake, 100-year flood, maximum wind gust, etc.
3. Extremely unlikely accidents	Less than once in 10,000 years but greater than once in 1,000,000 years	Accidents that probably will not occur during the life cycle of the facility; this includes a severe tornado, airplane crash, etc.
4. Not reasonably foreseeable accidents	Less than once in 1,000,000 years	All other accidents (e.g., a direct meteorite strike)

TE | a. Source: DOE (1994j).

This EIS evaluation examined a full spectrum of accidents; the tables in this appendix reflect the bounding (risk or consequence) event for the frequency ranges listed in Table E-1, in which risk is defined as the product of the frequency (events per year) and the consequence of an event.

The bounding consequence events would result in the largest projected increases in latent cancer fatalities, were these accidents to occur. The bounding risk events would represent the highest individual likelihood of contracting a fatal cancer, or the highest incremental cancer fatality rate in an exposed population, expressed in units of latent cancer fatalities per year. The tables in Section E.3 present the highest point estimate of risk to the maximally exposed offsite individual for each phase in bold type. This bolded number, when compared to the 1990 United States annual average risk of dying of cancer of about 0.002 (DOC 1992), provides a perspective on whether the event would be likely to increase an individual's lifetime cancer risk due to an accident dose received in that year.

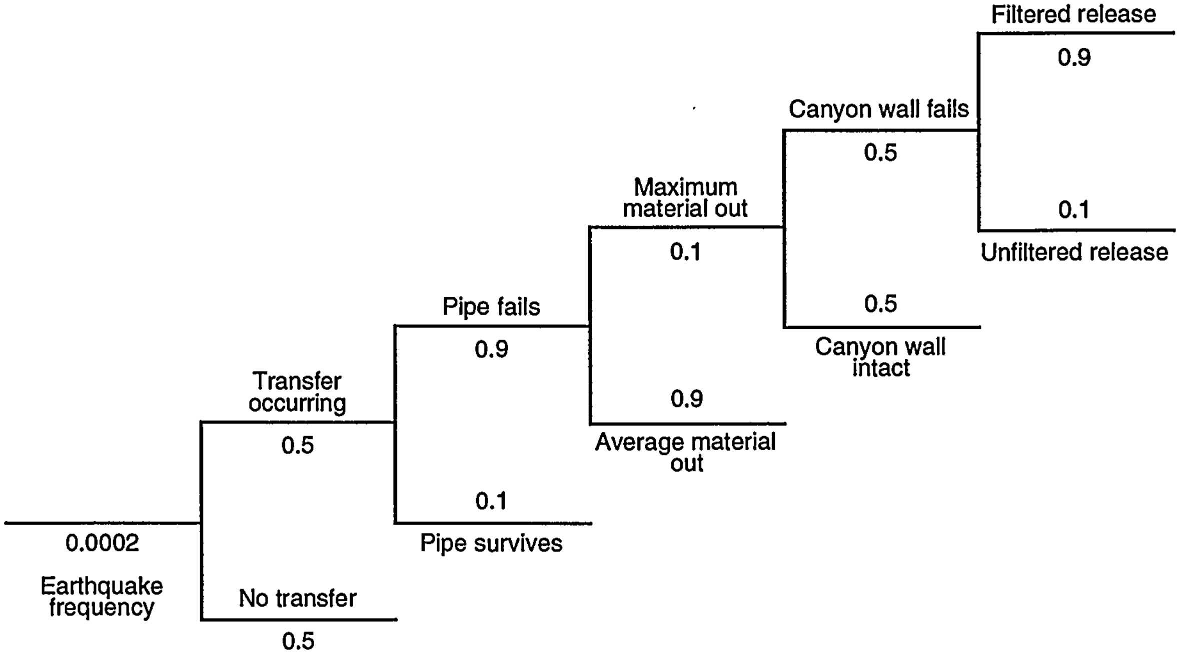
E.2 Accident Analysis Method

The accidents analyzed in this EIS would result from events that are considered "reasonably foreseeable" (expected to occur at least once in 1,000,000 years). The frequencies listed in the tables in this appendix are usually associated with the initial event that leads to a release of radioactive material. In most cases, this is a conservative frequency (i.e., it overestimates the risk) because in reality a chain of events, each with its own frequency, must occur; this includes the unlikely and highly unfavorable meteorological conditions assumed to prevail at the time of the accident. In addition, the analysis might have used conservative release assumptions to calculate potential consequences (doses) that could result from such accidents. These consequences are conservative because the release of radioactivity from the facility associated with the initiating event (e.g., earthquake) could occur only after the failure of a number of safety systems.

For example, a release of radioactive material from a chemical separations facility (e.g., F-Canyon) could occur in the following manner: An earthquake occurs during a tank-to-tank transfer of radioactive solution in the canyon. The transfer pipe fails or ruptures but the transfer continues and half the contents of the tank spill to the floor of the canyon. Simultaneous with the pipe rupture, the walls of the canyon crack, providing a release pathway to the environment. In addition, the canyon ventilation system fails. (The ventilation system normally maintains the interior of the canyon at a lower pressure than the outside environment. In this way, air leaks are normally into rather than out of the canyon.) After the radioactive material spills, a fraction becomes airborne and passes through the cracks in the canyon walls. This airborne radioactivity is blown off the Site.

This scenario is conservative because tank-to-tank transfers do not occur on a continuous basis, and the earthquake would have to occur during a transfer. DOE assumes that the following failures would allow the release to reach the offsite population at the projected dose levels: (1) the transfer pipe fails, (2) operators fail to respond or are unable to stop the transfer, (3) the canyon walls crack sufficiently to allow the escape of 10 percent of the airborne radioactive material, and (4) power distribution and electrical relays associated with the ventilation system fail. In addition, all released material escapes the facility in the first 2 hours and the meteorological conditions are such that only limited dispersion (or dilution) of the material has occurred by the time it reaches the SRS boundary. Figure E-1 is a sample event tree that shows the effects of this hypothetical earthquake.

The analytical method described in the following sections did not include emergency response actions to accident situations (e.g., evacuation of personnel to a safe distance or notification of the



For this example with these assigned frequencies, the estimated frequency for a maximum release exterior to the canyon would be (0.0002 × 0.5 × 0.9 × 0.1 × 0.5 × 0.9) or 0.000004; more than 50 times less likely than the frequency of the earthquake alone.

Figure E-1. Example of a fault tree.

public to perform such response actions as taking shelter) in its determination of potential impacts on workers or members of the public. To minimize potential human exposures and impacts on the environment from postulated accidents, the SRS has established an Emergency Plan (WSRC 1994f) that governs responses to potential accidents. TE

The presentation of data in this appendix uses an alternative scientific notation that facilitates comparisons of the results in tables that sometimes cover several pages. This notation is explained below:

7.1E-01 = 7.1×10^{-1}
or = 0.71

2.4E+3 = $2.4 \times 10^{+3}$
or = 2,400

The use of this notation shows the relative magnitude of any data entry. The absence of an "E" notation indicates an actual number without the need for a multiple of 10.

To approximate the potential accident impact contribution for each material (or group of materials) of interest, DOE created a flow diagram showing the location, condition, and chemical or physical form of the material. If a safety analysis report provided different frequencies for an event depending on location in a facility (e.g., a fire is more likely in a glovebox than in a dissolver tank), the analysis used the appropriate frequency for each location housing the material. In some cases, the current forms of the materials differ greatly, although two material groups both might contain primarily plutonium-239. At some point in the processing of both materials (under the preferred alternative), the original form would be lost and the newly generated form would be virtually identical. An example would be Mark-31 plutonium targets and H-Canyon plutonium solutions. After dissolution and processing, the Mark-31 targets would have formed "newly generated" plutonium solutions. These solutions would not pose the same level of concern as those in F-Canyon, which have been stored for several years longer than planned.

In addition to customizing the frequency by location, the analysis customized the source term and composition of the material to the extent possible. For example, if a solution has been processed through a canyon, concentrated, and purified by removing fission products, the source term was adjusted to the maximum concentration with fission product contributions subtracted. The effect of this type of customization is evident in the tables that list the impacts from earthquakes. The frequency remains constant, but both the quantity (in terms of curies) and the isotopic composition (e.g., more americium-241 than plutonium-239) vary by material. These variations enable discrimination of the impacts from one material to another. This discrimination can determine the potential risk reduction if the material of interest is stabilized.

If it was not possible to customize the frequency or source term for a material, this assessment used the results from the applicable safety analysis report. These results, which represent the bounding accident analysis, are useful for predictions of the impacts from a common mode failure (i.e., a severe TE earthquake). Table E-5 lists F-Canyon bounding severe earthquake impacts under the heading "F-Canyon (full operation)." Table E-6 lists H-Canyon bounding severe earthquake impacts under the heading "H-Canyon (limiting solution source term)." Severe earthquake impacts from the canyons would far exceed those from other facilities involved in the interim management of nuclear materials; a total impact due to a severe earthquake could be approximated by adding the individual impacts from F- and H-Canyons. This cumulative impact is conservative because it is unlikely that both canyons would experience the maximum effects from a severe earthquake.

E.2.1 AFFECTED FACILITIES

Appendix C discusses the facilities used for nuclear material management activities within the scope of this EIS. In addition to the primary areas that house nuclear material, other SRS facilities contain nuclear materials (e.g., the TNX facility has two tanks of depleted uranyl nitrate solution and N-Area has drums of depleted uranium oxide). DOE has evaluated these facilities for their potential hazards and has determined that safety analysis reports were not required due to the low hazards posed by the facilities. This means that a total release of materials without mitigation would result in consequences below the threshold requiring detailed analysis. As a result, the extent of quantitative impact data is limited. In most cases, the impacts will be compared to known impacts that bound those from secondary facilities. To determine the types of accident scenarios this appendix would present, DOE performed an extensive review of existing safety documentation for facilities that either perform or support activities that could be involved with management of nuclear material.

E.2.2 RADIOLOGICAL ACCIDENT ANALYSIS METHOD

DOE used computer models to determine the consequences resulting from the release of radioactivity. This evaluation assumed the release of 1 curie of pertinent isotopes to a surface stream (for liquids) or to the atmosphere at ground level and at an elevated level, such as through an exhaust stack for the various facilities involved in the alternatives discussed in this EIS. Using the computer models, the evaluation calculated doses to an uninvolved worker, the maximally exposed offsite individual, and the offsite population within 80 kilometers (50 miles) of the Site (Simpkins 1994a,b).

DOE used two SRS-specific computer codes -- AXAIR89Q and LADTAP XL -- to calculate the doses from each of the 1-curie releases postulated. Both codes perform accident analyses described in facility safety analysis reports and postulated accident impacts presented in other EISs developed for the SRS.

The AXAIR89Q computer code (WSRC 1994g), which was developed in accordance with guidelines TE established by the U.S. Nuclear Regulatory Commission for modeling atmospheric releases, models the doses from airborne constituents of postulated accidental releases of radionuclides to the environment. The modeling of the various accidents postulated for the facilities associated with the different alternatives assumed conservative (99.5 percentile) meteorological conditions (e.g., direction and speed of prevailing wind). "Conservative meteorological conditions" are those for which, for a given release, the concentration of radionuclides (and the resulting doses) at a fixed downwind location will not be exceeded 99.5 percent of the time. Usually, this means a highly stable-low wind speed weather condition where the wind provides only limited dilution of the material released. Use of these meteorological conditions results in consequences approximately three to four times higher for onsite workers and between 10 and 100 times higher for the offsite population than those that would occur during average (50 percentile) meteorological conditions.

The LADTAP XL computer code was developed to model aqueous (i.e., liquid) releases of radionuclides during routine operations and potential accidents. The modeling of the aqueous releases associated with the postulated accidents described in this appendix took no credit for the holdup of radionuclides within the soils surrounding the area where the accidents would occur. In other words, the modeling assumed that the entire release would discharge directly as a liquid to the ground, migrate to the Savannah River either directly or through Fourmile Branch, Pen Branch, etc., and enter the drinking water supply.

DOE calculated most of the impacts to individuals (e.g., dose equivalent, expressed as rem or TC projected cancer incidence) from postulated accidental releases of radionuclides to the environment for the various facilities by multiplying the quantity of each isotope in the source term release (in curies per isotope) presented in the safety analysis documents by the doses calculated for a 1-curie release, as discussed in the previous paragraphs. For example, if a facility safety analysis report stated that 0.00044 curie of strontium-90 was released at ground level in the F-Area, and the projected dose to the maximally exposed offsite individual from a 1-curie release of strontium-90 at ground level in the F-Area is 0.0001 rem, then the dose to the maximally exposed offsite individual from the release would be determined by multiplying 0.00044 curie by 0.0001 rem per curie, resulting in a dose of 0.000000044 (or 4.4×10^{-8}) rem. The total projected dose would equal the sum of the doses received from each radionuclide (isotope) released during the accident. This approach was not used for impacts already presented in NEPA documents (e.g., high-level waste tank accidents or Defense Waste Processing Facility accidents); in such cases, the impacts were obtained from those documents. Section E.3 presents the doses to uninvolved workers, maximally exposed offsite individuals, and the offsite population postulated for the facility radiological accidents evaluated in this appendix. TC

Each table in Section E.3 reflects the projected consequences in terms of dose (rem or person-rem), point estimate of risk (dose × frequency, in units of rem per year or person-rem per year), and latent cancer fatalities based on projections using guidelines developed by the International Commission of Radiation Protection (see Chapter 4). These guidelines, which are based on several decades of statistical analyses, provide a projection of an individual's chance of developing a cancer that proves to be fatal over time or a projection of the number of fatal cancers that would be likely to result from a population of individuals receiving a collective dose. These numbers enable comparisons of the highest consequence accidents among alternatives and among the phases of an alternative. The projections do not reflect the actual risk to an individual or population because the analysis does not consider the frequency of the accident (likelihood of occurrence). The risk of developing cancer resulting from SRS activities to manage nuclear material would be very low because accidents with large consequences from radioactive materials have not occurred historically and are unlikely to occur in the future. Each table also contains a column listing the total number of released curies estimated for each accident. The variations in dose estimates from similar release amounts is due to the varying impacts of different radioactive isotopes (e.g., 1 curie of plutonium-239 has almost five times the impact of 1 curie of plutonium-241).

As discussed above, this appendix describes risks to uninvolved workers and members of the public from radiological accidents involving nuclear materials in a quantitative fashion using such parameters as dose, accident frequency, and latent cancer fatalities in the population (as discussed in Section E.3). However, it presents potential impacts to involved, or "close-in" workers, from postulated accidents in a qualitative rather than a quantitative (in numerical terms) fashion, primarily because there is no adequate method for calculating meaningful consequences at or near the location where the accidental release occurs (DOE 1994h). The following example illustrates this concept. TE

A typical method for attempting to calculate the dose to an involved worker is to assume that the material is released in a room occupied by the worker and that the material instantly disperses throughout the room. Because the worker would be in the room when the release occurred, that individual probably would breathe some fraction of the radioactive materials for a number of seconds TE before leaving the room. Typically, estimates of exposure time are based on assumptions about worker response to the incident (e.g., how long before the worker left the room, or whether the worker evacuated the room through an area of higher airborne concentrations). For example, consider a hypothetical instance in which a worker dropped a container with 2,000 grams (4.4 pounds) of plutonium oxide powder. Depending on the size of the room where the release occurred, the assumptions made on how much of the released powder became airborne and respirable, and the length of time the exposed worker remained in the room, the calculable dose to the worker could be anywhere between 80 and 78,000 rem (DOE 1994h). The uncertainty of this estimate is large, and no additional insight into the activity is available because the occurrence is accepted as undesirable without needing to perform the calculations. Historic evidence (DOE 1994a) L12-2 suggests that this type of event would be a nonfatal accident resulting in room contamination with the potential for personnel contamination and assimilation. Presenting this wide range of worker dose is not helpful in comparisons of impacts among alternatives because the "overlap" would mask any discrimination. Section E.3.2 discusses potential radiological impacts to facility workers from accidents in a facility.

E.2.3 HAZARDOUS MATERIAL ACCIDENT ANALYSIS METHODOLOGY

A full understanding of the hazards associated with SRS nuclear facilities under the alternatives considered in this EIS requires analyses of potential accidents involving both hazardous and radiological materials. For chemically toxic materials, several government agencies recommend quantifying the health effects that cause short-term consequences as threshold values of concentrations in air. Because the long-term health consequences of human exposure to hazardous materials are not as well understood as those related to radiation exposure, a determination of potential health effects from exposures to hazardous materials is more subjective than a determination of health effects from exposure to radiation. Therefore, the consequences from accidents involving hazardous materials postulated in this appendix are in terms of airborne concentrations at various distances from the accident location, rather than dose or latent cancer fatalities. Because hazardous materials are used during the operations of each facility, the actual quantity associated with a particular alternative for the materials discussed in this EIS cannot be determined. For example, if a chemical is used to prevent microbiological growth in service water for a facility, then that chemical's tank or vessel must be assumed to be present for the duration of any facility function. Some or all of the hazardous substances could be eliminated if the mission of the facility were completed. None of the primary facilities involved in the storage or management of nuclear material is likely to complete its total mission within the period covered by this EIS.

To determine potential health effects to workers and members of the public that could result from accidents involving hazardous materials, DOE determined the airborne concentrations of such materials released during an accident where the uninvolved worker and offsite individual would be [i.e., 640 meters (2,100 feet) and the nearest SRS boundary, respectively] and compared them to the Emergency Response Planning Guideline (ERPG) values (AIHA 1991). The American Industrial Hygiene Association established these values, which depend on the material or chemical being considered, for three general severity levels to ensure that the necessary emergency actions occur to minimize worker and public exposures after accidents. These severity levels include the following:

- ERPG-1 Values. Exposure to airborne concentrations greater than ERPG-1 values for a period greater than 1 hour results in an unacceptable likelihood that a person would experience mild transient adverse health effects or perception of a clearly defined objectionable odor.

- ERPG-2 Values. Exposure to airborne concentrations greater than ERPG-2 values for a period greater than 1 hour results in an unacceptable likelihood that a person would experience or develop irreversible or other serious health effects or symptoms that could impair one's ability to take protective action.

- ERPG-3 Values. Exposure to airborne concentrations greater than ERPG-3 values for a period greater than 1 hour results in an unacceptable likelihood that a person would experience or develop life-threatening health effects.

Because all hazardous materials do not have ERPG values, DOE could not use such values to estimate potential impacts on the public from each hazardous material accident postulated for the SRS facilities discussed in this appendix. For chemicals that do not have ERPG values, this assessment compared airborne concentrations of hazardous materials resulting from postulated accidents to the most restrictive available exposure limits established by other guidelines (WSRC 1992b) to control worker exposures to hazardous materials. Table E-2 lists the hierarchy of exposure limits that DOE used to evaluate potential health effects resulting from postulated hazardous material accidents.

DOE used a bounding approach to determine the potential impacts on individuals at different positions (e.g., uninvolved workers and the maximally exposed offsite individual) from postulated accidents in each facility area from Extremely Hazardous Substances; the amounts of such substances

Table E-2. Hierarchy of established limits and guidelines used to determine impacts from postulated hazardous material accidents.

Primary airborne concentration guideline	Hierarchy of alternative guidelines (if primary guidelines are unavailable)	Reference of alternative guideline
ERPG-3	EEGL[a] (30-minute exposure)	NAS (1985)
	IDLH[b]	NIOSH (1990)
ERPG-2	EEGL (60-minute exposure)	NAS (1985)
	LOC[c]	EPA (1987)
	PEL-C[d]	29 CFR Part 1910.1000, Subpart Z
	TLV-C[e]	ACGIH (1992)
	TLV-TWA[f] multiplied by 5	ACGIH (1992
ERPG-1	TWA-STEL[g]	29 CFR Part 1910.100, Subpart Z
	TLV-STEL[h]	ACGIH (1992)
	TLV-TWA multiplied by 3	ACGIH (1992)

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a. Emergency Exposure Guidance Level (EEGL): "A concentration of a substance in air (as a gas, vapor, or aerosol) that may be judged by the Department of Defense to be acceptable for the performance of specific tasks during emergency conditions lasting for a period of 1 to 24 hours. Exposure at an EEGL might produce reversible effects that do not impair judgment and do not interfere with proper responses to an emergency." The EEGL is "...a ceiling guidance level for a single emergency exposure, usually lasting from 1 to 24 hours -- an occurrence expected to be infrequent in the lifetime of a person."

b. Immediately Dangerous to Life and Health (IDLH): "The maximum concentration from which, in the event of respirator failure, one could escape within 30 minutes without a respirator and without experiencing any escape-impairing (e.g., severe eye irritation) or irreversible health effects."

c. Level of Concern (LOC): "The concentration of an extremely hazardous substance in air above which there may be serious irreversible health effects or death as a result of a single exposure for a relatively short period of time."

d. Permissible Exposure Limit - Ceiling (C): "The employee's exposure which shall not be exceeded during any part of the work day."

e. Threshold Limit Value - Ceiling (TLV-C): "The concentration that should not be exceeded during any part of the working exposure."

f. Threshold Limit Value - Time Weighted Average (TLV-TWA): "The time-weighted average concentration for a normal 8-hour workday and a 40-hour workweek, to which nearly all workers may be repeatedly exposed, day after day, without adverse effect."

g. Time Weighted Average - Short-Term Exposure Limit (TWA-STEL): "The employee's 15-minute time weighted average exposure which shall not be exceeded at any time during a work day unless another time limit is specified...."

h. Threshold Limit Value - Short-Term Exposure Limit (TLV-STEL): "The concentration to which workers can be exposed continuously for a short period of time without suffering from (1) irritation, (2) chronic or irreversible tissue damage, or (3) narcosis of sufficient degree to increase the likelihood of accidental injury, impair self-rescue, or materially reduce work efficiency, and provided that the daily TLV-TWA is not exceeded."

and their locations were determined from the SRS Tier Two Emergency and Hazardous Chemical Inventory Report (WSRC 1994h). This annual report identifies the chemicals at the Site that are hazardous or that require the establishment of emergency response procedures. Following identification of the amounts and locations of the Extremely Hazardous Substances (see Section E.4) in each area, DOE calculated the airborne concentrations at 640 meters (2,100 feet) from the point of

TE

release and the nearest SRS boundary (i.e., locations of the uninvolved worker and maximally exposed offsite individual, respectively) that would be likely from a release of the maximum inventory of each Extremely Hazardous Substance in a single location. EPICode™ (Emergency Prediction and Information Code), a commercially available computer code for modeling routine or accidental releases of hazardous chemicals to the environment (Homann 1988), calculated the airborne concentrations at the different locations.

E.3 Postulated Accidents Involving Radioactive Materials

E.3.1 IMPACTS TO UNINVOLVED WORKERS AND MEMBERS OF THE PUBLIC

This EIS presents the consequences and risks of bounding accidents. In this EIS, the term "bounding accident" represents postulated events or accidents that have higher consequences or risks (i.e., consequences × frequencies) than other accidents postulated in the same frequency range. A consideration of the risks associated with bounding events or accidents for a facility can establish an understanding of the overall risk to workers, members of the public, and the environment from nuclear material management activities. In addition, the risks of different alternatives can be compared relatively by comparing the risks associated with the bounding accidents for the phases of each alternative. Figure E-2 shows the concept of bounding risk accidents. The accident impact tables in this section list the bounding events for each pertinent frequency range. These tables list in bold type the highest overall point estimate of risk for the maximally exposed offsite individual and the highest consequence to the population for each phase. Some tables also list a representative selection from the full spectrum of accidents to aid in comparisons among alternatives or to demonstrate the elimination of some accidents for specific materials.

Table E-3 is a summary matrix of the facilities used for each phase of the alternatives considered for each material category. The No-Action Alternative column lists the facility where the material is currently stored; this alternative has no phases. The "conversion" phase refers to any initial treatment; it is not limited to processing in a canyon. Not all alternatives have all phases (e.g., there could be only one conversion phase; therefore, the interim storage and additional conversion phases would not be applicable).

Table E-3 is intended for use in conjunction with Tables E-4 through E-12, which list accident analysis data for each material and the facilities that could be involved in a specific phase for the corresponding material. Table E-3 can be used to determine the facility accidents analyzed that

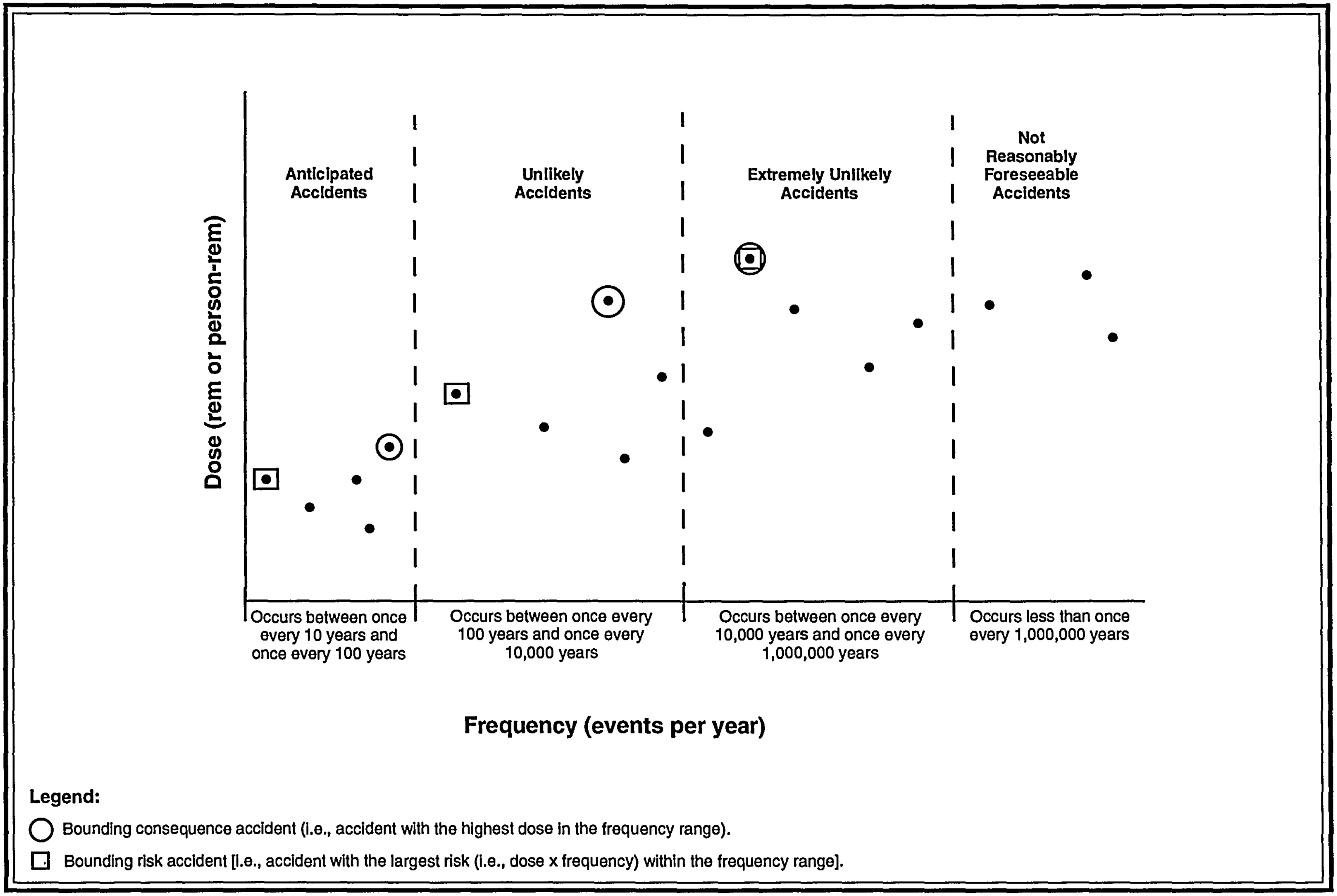

Figure E-2. Methodology used to determine bounding risk accidents for the various nuclear facilities.

Table E-3. Applicable facilities for each alternative.

Material (Table)	No-Action Alternative	Other alternatives	Conversion	Interim storage	Additional conversion	Post-stabilization storage
Plutonium-242 (Table E-4)	H-Canyon[a]	Oxide	H-Canyon[a] HB-Line	Not applicable	Not applicable	Existing vaults[b]
		Vitrify	H-Canyon[a] HB-Line	Existing vaults[b]	FB-Line Recovery F-Canyon	(c)
		Waste	H-Canyon[a]	High-level waste[d]	DWPF[d]	(c)
		Metal	Solution transport (Section 4.2.2)	F-Canyon	FB-Line processing Actinide Packaging and Storage Facility	Storage vault[b]
Americium/curium (Table E-5)	F-Canyon	Vitrify	F-Canyon	Not applicable	Not applicable	(c)
		Waste	F-Canyon	High-level waste[d]	DWPF[d]	(c)
		Oxide	F-Canyon F-Canyon hot cell[e]	Not applicable	Not applicable	Existing vault[b]
Neptunium (Table E-6)	H-Canyon	Oxide	H-Canyon HB-Line	Existing vaults[b]	Actinide Packaging and Storage Facility[f]	Storage vault[b]
		Vitrify	Solution transport (Section 4.2.2)	F-Canyon	F-Canyon	(c)
		Waste	H-Canyon	High-level waste[d]	DWPF[d]	(c)
H-Canyon plutonium-239 solutions (Table E-7)	H-Canyon	Oxide	H-Canyon HB-Line	Existing vaults[b]	Actinide Packaging and Storage Facility[f]	Storage vault[b]
		Liquid waste (DWPF)[g]	H-Canyon	High-level waste[d]	DWPF[d]	(c)
		Vitrify	Solution transport (Section 4.2.2)	F-Canyon	F-Canyon	(c)
		Metal	Solution transport (Section 4.2.2)	F-Canyon	FB-Line Actinide Packaging and Storage Facility	Storage vault[b]
H-Canyon enriched uranium solutions (Table E-8)	H-Canyon H-Outside	Oxide (low enriched uranium)	FA-Line	Not applicable	Not applicable	Storage vault[h]
		Oxide (enriched uranium)	Uranium Solidification Facility	Not applicable	Not applicable	Storage vault[h]
		Liquid waste (DWPF)	H-Canyon	High-level waste[d]	DWPF[d]	(c)

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Table E-3. (continued).

Material (Table)	No-Action Alternative	Other alternatives	Conversion	Interim storage	Additional conversion	Post-stabilization storage
Plutonium and uranium stored in vaults (Table E-9)	235-F FB-Line	Metal	HB-Line Phase I H-Canyon (limiting) HB-Line Phase II	Existing vaults[b]	FB-Line Actinide Packaging and Storage Facility[f]	Storage vault[b]
		Oxide	HB-Line Phase I (including Pu-238) H-Canyon (Frame waste) HB-Line Phase II HB-Line Phase III	Existing vaults[b]	Actinide Packaging and Storage Facility[f]	Storage vault[b]
		Repackage	Actinide Packaging and Storage Facility	Not applicable	Not applicable	Storage vault[b]
		Liquid waste (DWPF)	HB-Line Phase I (including Pu-238)	High-level waste[d]	DWPF[d]	(c)
		Vitrify	HB-Line Phase I H-Canyon (limiting) HB-Line Phase II	Existing vaults[b]	F-Canyon	(c)
Mk-31 targets (Table E-10)	L-Reactor Basin	Metal	F-Canyon FA-Line FB-Line	Existing vaults[b]	Actinide Packaging and Storage Facility[g]	Storage vault[b]
		Liquid waste (DWPF)	F-Canyon	High-level waste[d]	DWPF[d]	(c)
		Dry Storage	Transfer to dry storage facility	Not applicable	Not applicable	Dry storage facility
		Vitrify	F-Canyon FA-Line FB-Line	Existing vaults[b]	F-Canyon	(c)
		Oxide	F-Canyon FA-Line FB-Line	Existing vaults[b]	Actinide Packaging and Storage Facility	Storage vault[b]
Mk-16/22 fuels (Table E-11)	Reactor basins	Oxide (low enriched uranium)	F/H-Canyon[i] H-Outside FA-Line	Not applicable	Not applicable	Storage vault[h]
		Dry storage[j]	Transfer to dry storage facility	Not applicable	Not applicable	Dry storage facility
		Oxide (enriched uranium)	H-Canyon H-Outside Uranium Solidification Facility	Not applicable	Not applicable	Storage vault[h]
		Liquid waste (DWPF)	F/H-Canyon[i] F/H-Outside	High-level waste[d]	DWPF[d]	(c)

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Table E-3. (continued).

Material (Table)	No-Action Alternative	Other alternatives	Conversion	Interim storage	Additional conversion	Post-stabilization storage
Other aluminum-clad targets[k]	Bounded by Mk-31 No-Action	Liquid waste (DWPF)	Bounded by Mk-16/22 liquid waste alternative (see Table E-11)			
	(See Table E-10) or Mk-16/22 (See Table E-11)	Dry storage	Bounded by Mk-16/22 dry storage alternative (See Table E-11)			
TRR and EBR-II fuel (Table E-12)	Receiving Basin for Offsite Fuels	Metal	F-Canyon FA-Line FB-Line	Existing vaults[b]	Actinide Packaging and Storage Facility[f]	Storage vault[b]
		Liquid waste (DWPF)	F-Canyon	High-level waste[d]	DWPF[d]	(c)
		Dry Storage	Transfer to Dry Storage Facility	Not applicable	Not applicable	Dry Storage Facility
		Vitrify	F-Canyon FA-Line FB-Line	Existing vaults[b]	F-Canyon	(c)
		Oxide	F-Canyon FA-Line FB-Line	Existing vaults[b]	Actinide Packaging and Storage Facility	Storage vault[b]

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a. The accident analysis for F-Canyon was used for plutonium-242 alternatives because it is more representative of this solution's source term.

b. Accident analysis for the 235-F facility is representative for both existing and new storage vaults; for new storage vaults, the analysis assumes that the ruptured storage container accident would not be credible after repackaging and improving storage conditions.

c. No credible accidents resulting in a release from vitrified material have been postulated.

d. Accident analysis information for the existing/projected facility inventory; if this information requires revision after analysis for different *isotopic* content, safety documentation will be updated in accordance with DOE Orders 5480.23 and 5480.21.

e. The americium/curium source term was used in the relevant accident scenarios for HB-Line to provide a representative accident analysis for the americium/curium Processing to Oxide Alternative.

f. The source terms associated with FB-Line drying are used in conjunction with FB-Line accidents to be representative of the new Actinide Packaging and Storage Facility.

g. DWPF = Defense Waste Processing Facility.

h. Accident analysis for storage operations at the Uranium Solidification Facility are representative for new uranium storage vaults.

i. This alternative enables either canyon to process fuel; H-Canyon accidents are representative for Mk-16 and -22 processing.

j. Table E-11 lists values that represent both dry storage and accelerated dry storage.

k. Because this material group consists of small quantities of a wide variety of aluminum-clad fuels, the accident impacts from this material group would be minimal. Each alternative for this material group is bounded by the accident analysis presented for other groups. Therefore, impacts reference the bounding accident analysis.

Table E-4. Plutonium-242.

Accident	Quantity released (curies)	Frequency (per year)	Accident consequences: Uninvolved worker (rem)	Accident consequences: MEI[a] (rem)	Accident consequences: Offsite population (person-rem)	Latent cancer fatalities (LCF) (Point estimate of increased risk per year) (Increased risk of LCF per occurrence): Uninvolved worker	Latent cancer fatalities (LCF): MEI	Latent cancer fatalities (LCF): Offsite population
NO ACTION								
H-Canyon (without dissolver)								
Airborne release of plutonium solution resulting from coil and tube failure in F-Canyon water cooling tower	17.0	4.00E-02	16.5	0.755	4.42E+03	2.6E-04 6.6E-03	1.5E-05 3.8E-04	8.8E-02 2.2
Severe earthquake	64.7	2.00E-04	9.91	0.447	2.64E+03	7.9E-07 4.0E-03	4.5E-08 2.2E-04	2.6E-04 1.3
Fire in a plutonium process vessel	56.2	6.10E-05	10.6	1.75	1.29E+04	2.6E-07 4.2E-03	5.3E-08 8.8E-04	3.9E-04 6.5
Inadvertent transfer of plutonium solution from a processing vessel to the ground outside building	24.9	7.40E-05	1.61	7.24E-02	4.30E+02	4.8E-08 6.4E-04	2.7E-09 3.6E-05	1.6E-05 0.22
CONVERSION								
H-Canyon (without dissolver)								
Same accident analysis as that for the No-Action Alternative								
HB-Line, Phase II (normal processing)								
Severe earthquake	7.00E-04	2.00E-04	1.79E-02	8.28E-04	4.83	1.4E-09 7.2E-06	8.3E-11 4.1E-07	4.8E-07 2.4E-03
Unpropagated fire in gloveboxes	1.60E-03	4.70E-02	6.46E-03	1.07E-03	7.88	1.2E-07 2.6E-06	2.5E-08 5.4E-07	1.9E-04 3.9E-03

TC

TC

Table E-4. (continued).

Accident	Quantity released (curies)	Frequency (per year)	Accident consequences: Uninvolved worker (rem)	Accident consequences: MEI[a] (rem)	Accident consequences: Offsite population (person-rem)	Latent cancer fatalities (LCF): Uninvolved worker	Latent cancer fatalities (LCF): MEI	Latent cancer fatalities (LCF): Offsite population
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
INTERIM STORAGE								
Existing Vaults (235-F)								
Rupture storage container (e.g., radiolytic decay)	5.14E-04	2.00E-02	8.62E-04	1.43E-04	1.05	6.9E-09 3.4E-07	1.4E-09 7.2E-08	1.1E-05 5.3E-04
Severe earthquake	1.05E-02	2.00E-04	0.60	7.0E-03	10	4.8E-08 2.4E-04	7.0E-10 3.5E-06	1.0E-06 5.0E-03
Fire	2.0E-5	5.0E-02	6.0E-04	2.0E-05	0.10	1.2E-08 2.4E-07	5.0E-10 1.0E-08	2.5E-06 5.0E-05
High-Level Waste Tanks								
Severe earthquake	(b)	2.00E-04	(b)	3.41E-03	0.26	(b) (b)	3.4E-10 1.7E-06	2.6E-08 1.3E-04
Hydrogen explosion in a tank	(b)	2.00E-05	0.291	1.13E-02	0.43	2.3E-09 1.2E-04	1.1E-10 5.7E-06	4.3E-09 2.2E-04
Waste tank filter fire	(b)	2.5E-02	9.55E-02	3.68E-03	8.5	9.6E-07 3.8E-05	4.6E-08 1.8E-06	1.1E-04 4.3E-03
F-Canyon (without dissolver)								
Airborne release of plutonium solution resulting from coil and tube failure in F-Canyon water cooling tower	17.0	4.00E-02	16.5	0.755	4.42E+03	2.6E-04 6.6E-03	1.5E-05 3.8E-04	8.8E-02 2.2
Severe earthquake	64.7	2.00E-04	9.91	0.446	2.64E+03	7.9E-07 4.0E-03	4.5E-08 2.2E-04	2.6E-04 1.3
Fire in a plutonium process vessel	56.2	6.10E-05	10.6	1.75	1.29E+04	2.6E-07 4.3E-03	5.3E-08 8.8E-04	3.9E-04 6.5
Inadvertent transfer of plutonium solution from a processing vessel to the ground outside building	24.9	7.40E-05	1.61	7.24E-02	4.30E+02	4.8E-08 6.4E-04	2.7E-09 3.6E-05	1.6E-05 0.22

TC

Table E-4. (continued).

Accident	Quantity released (curies)	Frequency (per year)	Accident consequences: Uninvolved worker (rem)	Accident consequences: MEI[a] (rem)	Accident consequences: Offsite population (person-rem)	Latent cancer fatalities (LCF): Uninvolved worker	Latent cancer fatalities (LCF): MEI	Latent cancer fatalities (LCF): Offsite population	
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)			
ADDITIONAL CONVERSION									
FB-Line (Recovery Operations)									
Severe earthquake	0.434	2.00E-04	1.13	5.02E-02	3.06E+02	9.0E-08 4.5E-04	5.0E-09 2.5E-05	3.1E-05 1.5E-01	
Inadvertent nuclear criticality in processing solution or solid	(a)	1.40E-04	(a)	2.64E-03	2.93	(a) (a)	1.8E-10 1.3E-06	2.1E-07 1.5E-03	
Propagated fire in processing vessels or gloveboxes	4.31E-04	5.26E-03	1.78E-03	2.92E-05	0.216	3.7E-09 7.1E-07	7.7E-11 1.5E-08	5.7E-07 1.1E-04	
F-Canyon (Second Pu Cycle Pu Contribution)									
Airborne release Pu solution resulting from coil & tube failure in F-Area Canyon water cooling tower	0.218	4.00E-02	0.531	2.44E-02	1.44E+02	8.8E-06 2.2E-04	**4.8E-07** 1.2E-05	2.9E-03 7.2E-02	
Severe earthquake	0.365	2.00E-04	3.43	0.158	9.22E+02	2.8E-07 1.4E-03	1.6E-08 7.9E-05	9.2E-05 0.46	
Fire in a plutonium process vessel	1.59	6.10E-05	2.27	0.378	2.78E+03	5.5E-08 9.0E-04	1.2E-08 1.9E-04	8.5E-05 1.4	TC
Inadvertent transfer of plutonium solution from a processing vessel to the ground outside building	9.65E-02	7.40E-05	0.872	4.02E-02	2.35E+02	2.6E-08 3.5E-04	1.5E-09 2.0E-05	8.7E-06 0.12	
Defense Waste Processing Facility									
Uncontrolled reaction	(b)	4.50E-02	1.50E-03	1.70E-04	2.50	2.70E-08 6.00E-07	3.83E-09 8.50E-08	5.63E-05 1.25E-03	
Meter spill	(b)	9.30E-03	2.94E-01	3.40E-02	4.90E+02	1.09E-06 1.18E-04	1.58E-07 1.70E-05	2.28E-03 2.45E-01	TC
Earthquake	(b)	5.20E-05[c]	4.04E+03	6.77	7.60E+04	(d) (d)	1.76E-07 3.38E-03	1.98E-03 **3.80E+01**	

Table E-4. (continued).

			Accident consequences			Latent cancer fatalities (LCF)		
Accident	Quantity released (curies)	Frequency (per year)	Uninvolved worker (rem)	MEI[a] (rem)	Offsite population (person-rem)	Uninvolved worker (Point estimate of increased risk per year) (Increased risk of LCF per occurrence)	MEI	Offsite population
			METAL					
FB-Line (processing)								
Severe earthquake	4.34	2.00E-04	11.3	0.521	3.06E+03	9.0E-07 4.5E-03	5.2E-08 2.6E-04	3.0E-04 1.5
Inadvertent nuclear criticality in processing solution or solid	(b)	1.40E-04	(b)	2.64E-03	2.93	(b) (b)	1.8E-10 1.3E-06	2.1E-07 1.5E-03
Propagated fire in processing vessels or gloveboxes	0.105	5.26E-03	4.33E-02	7.13E-03	52.7	9.1E-08 1.7E-05	1.9E-08 1.3E-06	1.4E-04 2.6E-02
Actinide Packaging and Storage Facility (FB-Line drying)								
Severe earthquake	1.74	2.00E-04	4.54	0.208	1.22E+03	3.6E-07 1.8E-03	2.0E-08 1.0E-04	1.2E-04 0.62
Inadvertent nuclear criticality	(b)	5.26E-05	(b)	2.64E-03	2.93	(b) (b)	6.9E-11 1.3E-06	7.7E-08 1.5E-03
Propagated fire in gloveboxes	3.37E-03	5.26E-03	1.39E-03	2.29E-04	1.69	2.9E-09 5.5E-07	6.1E-10 1.2E-07	4.5E-06 8.4E-04
			POST-STABILIZATION STORAGE					
Existing Vaults (235-F)								
Rupture storage container (e.g., radiolytic decay)	5.14E-04	2.00E-02	8.62E-04	1.43E-04	1.05	6.9E-09 3.4E-07	**1.4E-09** 7.2E-08	1.1E-05 5.3E-04
Severe earthquake	1.05E-02	2.00E-04	0.60	7.0E-03	10	4.8E-08 2.4E-04	7.0E-10 3.5E-06	1.0E-06 **5.0E-03**
Fire	2.0E-05	5.0E-02	6.0E-04	2.0E-05	0.10	1.2E-08 2.4E-04	5.0E-10 3.5E-06	2.5E-06 5.0E-05

TC

Table E-4. (continued).

			Accident consequences			Latent cancer fatalities (LCF)		
Accident	Quantity released (curies)	Frequency (per year)	Uninvolved worker (rem)	MEI[a] (rem)	Offsite population (person-rem)	Uninvolved worker	MEI	Offsite population
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
			POST-STABILIZATION STORAGE (continued)					
Storage Vault								
Severe earthquake	1.05E-02	2.00E-04	0.60	7.0E-03	10	4.8E-08 2.4E-04	7.0E-10 3.5E-06	1.0E-06 **5.0E-03**
Fire	2.0E-05	5.0E-02	6.0E-04	2.0E-05	0.10	1.2E-08 2.4E-07	5.0E-10 1.0E-08	2.5E-06 5.0E-05

TC

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a. MEI = maximally exposed individual.
b. These data were not available.
c. This is the frequency due to the postulated sequence of events; it is based on earthquake frequency of 0.0002 event per year.
d. The number of latent cancer fatalities is not calculated because the dose (4,000 rem) would result in death within a few days.

Table E-5. Americium and curium. | TE

Accident	Quantity released (curies)	Frequency (per year)	Accident consequences: Uninvolved worker (rem)	Accident consequences: MEI[a] (rem)	Accident consequences: Offsite population (person-rem)	Latent cancer fatalities (LCF): Uninvolved worker	Latent cancer fatalities (LCF): MEI	Latent cancer fatalities (LCF): Offsite population	
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)			
NO ACTION									
F-Canyon (americium/curium solutions only)									
Severe earthquake[b]	0.360	2.00E-04	5.39	0.241	1.43E+03	4.31-07 2.2E-03	2.4E-08 1.2E-04	1.4E-04 0.72	
Inadvertent transfer of americium/curium solution to F-Canyon sump	1.01E-02	3.30E-02	2.12E-02	3.47E-03	26.0	2.8E-07 8.5E-06	**5.7E-08** 1.7E-06	4.3E-04 1.3E-02	
Inadvertent transfer of americium/curium solution from processing vessel to ground outside building	1.73	8.80E-05	23.1	1.04	6.11E+03	1.6E-06 1.8E-02	4.6E-08 5.2E-04	2.7E-04 **3.1**	
CONVERSION									
F-Canyon (full operation)									
Airborne release of solution resulting from coil and tube failure in F-Canyon water cooling tower	17.0	4.00E-02	16.5	0.755	4.42E+03	2.6E-04 6.6E-03	**1.5E-05** 3.8E-04	8.8E-02 2.2	TC
Severe earthquake	73.0	2.00E-04	10.5	0.474	2.80E+03	8.4E-07 4.2E-03	4.7E-08 2.4E-04	2.8E-04 1.4	
Fire in process vessel	56.2	6.10E-05	10.6	1.75	1.29E+04	2.6E-07 4.2E-03	5.3E-08 8.8E-04	3.9E-04 **6.5**	
Inadvertent transfer of solution from a processing vessel to the ground outside building	24.9	1.10E-04	1.61	7.24E-02	4.30E+02	7.7E-08 6.4E-04	4.0E-09 3.6E-05	2.4E-05 0.22	TC

Table E-5. (continued).

			Accident consequences			Latent cancer fatalities (LCF)		
Accident	Quantity released (curies)	Frequency (per year)	Uninvolved worker (rem)	MEI[a] (rem)	Offsite population (person-rem)	Uninvolved worker	MEI	Offsite population
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
CONVERSION (continued)								
F-Canyon hot cell (americium/curium line)								
Severe earthquake	0.48	2.00E-04	6.1	0.28	1.6E+03	4.9E-07	2.8E-08	1.6E-04
						2.4E-03	1.4E-04	0.80
Unpropagated fire in gloveboxes	1.1	4.70E-02	2.2	0.36	2.7E+03	4.1E-05	8.5E-06	6.3E-02
						8.8E-04	1.8E-04	1.4
INTERIM STORAGE								
High-Level Waste Tanks								
Severe earthquake	(c)	2.00E-04	(c)	3.41E-03	0.26	(c)	3.4E-10	2.6E-08
						(c)	1.7E-06	1.3E-04
Hydrogen explosion in a tank	(c)	2.00E-05	0.291	1.13E-02	0.43	2.3E-09	1.1E-10	4.3E-09
						1.2E-04	5.7E-06	2.2E-04
Waste tank filter fire	(c)	2.5E-02	9.55E-02	3.68E-03	8.5	9.6E-07	**4.6E-08**	1.1E-04
						3.8E-05	1.8E-06	**4.3E-03**
ADDITIONAL CONVERSION								
Defense Waste Processing Facility								
Uncontrolled reaction	(c)	4.50E-02	1.50E-03	1.70E-04	2.50	2.70E-08	3.83E-09	5.63E-05
						6.00E-07	8.50E-08	1.25E-03
Melter spill	(c)	9.30E-03	2.94E-01	3.40E-02	4.90E+02	1.09E-06	1.58E-07	2.28E-03
						1.18E-04	1.70E-05	2.45E-01
Earthquake	(c)	5.20E-05[d]	4.04E+03	6.77	7.60E+04	(e)	**1.76E-07**	1.98E-03
						(e)	3.38E-03	**3.80E+01**

TC

Table E-5. (continued).

			Accident consequences			Latent cancer fatalities (LCF)		
Accident	Quantity released (curies)	Frequency (per year)	Uninvolved worker (rem)	MEI[a] (rem)	Offsite population (person-rem)	Uninvolved worker (Point estimate of increased risk per year) (Increased risk of LCF per occurrence)	MEI (Point estimate of increased risk per year) (Increased risk of LCF per occurrence)	Offsite population (Point estimate of increased risk per year) (Increased risk of LCF per occurrence)
POST-STABILIZATION STORAGE								
Existing Vault (235-F)								
Rupture of storage container (e.g., radiolytic decay)	5.14E-04	2.00E-02	8.62E-04	1.43E-04	1.05	6.9E-09 3.4E-07	**1.4E-09** 7.2E-08	1.1E-05 5.3E-04
Severe earthquake	1.05E-02	2.00E-04	0.60	7.0E-03	10	4.8E-08 2.4E-04	7.0E-10 3.5E-06	1.0E-06 **5.0E-03**
Fire	2.0E-05	5.0E-02	6.0E-04	2.0E-05	0.10	1.2E-08 2.4E-07	5.0E-10 1.0E-08	2.5E-06 5.0E-05

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a. MEI = Maximally exposed individual.
b. Contribution from americium/curium only, not entire contents of F-Canyon.
c. These data were not available.
d. This is the frequency due to the postulated sequence of events, it is based on earthquake frequency of 0.0002 event per year.
e. The number of latent cancer fatalities is not calculated because the dose (4,000 rem) would result in death within a few days.

TC

Table E-6. Neptunium.

TE

Accident	Quantity released (curies)	Frequency (per year)	Accident consequences: Uninvolved worker (rem)	Accident consequences: MEI[a] (rem)	Accident consequences: Offsite population (person-rem)	Latent cancer fatalities (LCF): Uninvolved worker	Latent cancer fatalities (LCF): MEI	Latent cancer fatalities (LCF): Offsite population
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
NO ACTION								
H-Canyon (limiting solution source term)								
Unpropagated fire in solution vessel	0.594	2.02E-02	2.15	0.355	2.62E+03	1.7E-05 8.6E-04	**3.6E-06** 1.8E-04	2.6E-02 1.3
Inadvertent transfer from a processing vessel to the ground outside the H-Canyon building	1.32	4.00E-04	31.0	1.42	8.27E+03	1.0E-05 2.4E-02	2.8E-07 7.1E-04	1.7E-03 **4.1**
Inadvertent transfer of solution to H-Canyon sump	3.15E-02	8.10E-02	0.114	1.88E-02	1.39E+02	3.7E-06 4.6E-05	7.6E-07 9.4E-06	5.6E-03 7.0E-02
Airborne release of solutions resulting from coil and tube failure in H-Canyon cooling system	2.08E-02	2.55E-03	0.136	6.25E-03	36	1.4E-07 5.4E-05	8.0E-09 3.1E-06	4.6E-05 1.8E-02
Severe earthquake	1.17	2.00E-04	27.4	1.26	7.31E+03	4.4E-06 2.2E-02	1.3E-07 6.3E-04	7.3E-04 3.7
CONVERSION								
H-Canyon (limiting solution source term)								
Same accident analysis as that for the No-Action Alternative								
HB-Line, Phase II (normal processing)								
Severe earthquake	7.00E-04	2.00E-04	1.79E-02	8.28E-04	4.83	1.4E-09 7.2E-06	8.3E-11 4.1E-07	4.8E-07 2.4E-03
Unpropagated fire in gloveboxes	1.60E-03	4.70E-02	6.46E-03	1.07E-03	7.88	1.2E-07 2.6E-06	2.5E-08 5.4E-07	1.9E-04 3.9E-03

Table E-6. (continued).

Accident	Quantity released (curies)	Frequency (per year)	Accident consequences: Uninvolved worker (rem)	Accident consequences: MEI[a] (rem)	Accident consequences: Offsite population (person-rem)	Latent cancer fatalities (LCF): Uninvolved worker	Latent cancer fatalities (LCF): MEI	Latent cancer fatalities (LCF): Offsite population	
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)			
INTERIM STORAGE									
Existing Vaults (235-F)									
Rupture storage container (e.g., radiolytic decay)	5.14E-04	2.00E-02	8.62E-04	1.43E-04	1.05	6.9E-09 3.4E-07	1.4E-09 7.2E-08	1.1E-05 5.3E-04	
Severe earthquake	1.05E-02	2.00E-04	0.60	7.0E-03	10	4.8E-08 2.4E-04	7.0E-10 3.5E-06	1.0E-06 5.0E-03	
Fire	2.0E-5	5.0E-02	6.0E-04	2.0E-05	0.10	1.2E-08 2.4E-07	5.0E-10 1.0E-08	2.5E-06 5.0E-05	
F-Canyon (without dissolver)									
Airborne release of solution resulting from coil and tube failure in F-Canyon water cooling tower	17.0	4.00E-02	16.5	0.755	4.42E+03	2.6E-04 6.6E-03	**1.5E-05** 3.8E-04	8.8E-02 2.2	TC
Severe earthquake	64.7	2.00E-04	9.91	0.447	2.64E+03	7.9E-07 4.0E-03	4.5E-08 2.2E-04	2.6E-04 1.3	
Fire in a plutonium process vessel	56.2	6.10E-05	10.6	1.75	1.29E+04	2.6E-07 4.3E-03	5.3E-08 8.8E-04	3.9E-04 **6.5**	
Inadvertent transfer of solution from a processing vessel to the ground outside building	24.9	7.40E-05	1.61	7.24E-02	4.30E+02	4.8E-08 6.4E-04	2.7E-09 3.6E-05	1.6E-05 0.22	TC
High-Level Waste Tanks									
Severe earthquake	(b)	2.00E-04	(b)	3.41E-03	0.26	(b) (b)	3.4E-10 1.7E-06	2.6E-08 1.3E-04	
Hydrogen explosion in a tank	(b)	2.00E-05	0.291	1.13E-02	0.43	2.3E-09 1.2E-04	1.1E-10 5.7E-06	4.3E-09 2.2E-04	TC
Waste tank filter fire	(b)	2.5E-02	9.55E-02	3.68E-03	8.5	9.6E-07 3.8E-05	4.6E-08 1.8E-06	1.1E-04 4.3E-03	

Table E-6. (continued).

			Accident consequences			Latent cancer fatalities (LCF)			
Accident	Quantity released (curies)	Frequency (per year)	Uninvolved worker (rem)	MEI[a] (rem)	Offsite population (person-rem)	Uninvolved worker	MEI	Offsite population	
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)			
ADDITIONAL CONVERSION									
Actinide Packaging and Storage Facility (FB-Line drying)									
Severe earthquake	1.74	2.00E-04	4.54	0.208	1.22E+03	3.6E-07 1.8E-03	2.0E-08 1.0E-04	1.2E-04 0.62	TC
Propagated fire in gloveboxes	3.37E-03	5.26E-03	1.39E-03	2.29E-04	1.69	2.9E-09 5.5E-07	6.1E-10 1.2E-07	4.5E-06 8.4E-04	
F-Canyon (second plutonium cycle contribution)									
Airborne release solution resulting from coil and tube failure in F-Canyon water cooling tower	0.218	4.00E-02	0.531	2.44E-02	1.44E+02	8.8E-06 2.2E-04	**4.8E-07** 1.2E-05	2.9E-03 7.2E-02	
Severe earthquake	0.365	2.00E-04	3.43	0.158	9.22E+02	2.8E-07 1.4E-03	1.6E-08 7.9E-05	9.2E-05 0.46	
Fire in a process vessel	1.59	6.10E-05	2.27	0.378	2.78E+03	5.5E-08 9.0E-04	1.2E-08 1.9E-04	8.5E-05 1.4	TC
Inadvertent transfer of solution from a processing vessel to the ground outside building.	9.65E-02	7.40E-05	0.872	4.02E-02	2.35E+02	2.6E-08 3.5E-04	1.5E-09 2.0E-05	8.7E-06 0.12	
Defense Waste Processing Facility									
Uncontrolled reaction	(b)	4.50E-02	1.50E-03	1.70E-04	2.50	2.70E-08 6.00E-07	3.83E-09 8.50E-08	5.63E-05 1.25E-03	
Melter spill	(b)	9.30E-03	2.94E-01	3.40E-02	4.90E+02	1.09E-06 1.18E-04	1.58E-07 1.70E-05	2.28E-03 2.45E-01	TC
Earthquake	(b)	5.20E-05[c]	4.04E+03	6.77	7.60E+04	(d) (d)	1.76E-07 3.38E-03	1.98E-03 **3.80E+01**	

Table E-6. (continued).

Accident	Quantity released (curies)	Frequency (per year)	Accident consequences: Uninvolved worker (rem)	Accident consequences: MEI[a] (rem)	Accident consequences: Offsite population (person-rem)	Latent cancer fatalities (LCF): Uninvolved worker	Latent cancer fatalities (LCF): MEI	Latent cancer fatalities (LCF): Offsite population
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
POST-STABILIZATION STORAGE								
Storage Vault								
Severe earthquake	1.05E-02	2.00E-04	0.60	7.0E-03	10	4.8E-08	**7.0E-10**	1.0E-06
						2.4E-04	3.5E-06	**5.0E-03**
Fire	2.0E-05	5.0E-02	6.0E-04	2.0E-05	0.10	1.2E-08	5.0E-10	2.5E-06
						2.4E-07	1.0E-08	5.0E-05

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a. MEI = maximally exposed individual.

b. These data were not available.

c. This is the frequency due to the postulated sequence of events; it is based on earthquake frequency of 0.0002 event per year.

d. The number of latent cancer fatalities is not calculated because the dose (4,000 rem) would result in death within a few days.

TC

Table E-7. H-Canyon plutonium-239 solutions.

TE

Accident	Quantity released (curies)	Frequency (per year)	Accident consequences			Latent cancer fatalities (LCF)		
			Uninvolved worker (rem)	MEI^a (rem)	Offsite population (person-rem)	Uninvolved worker	MEI	Offsite population
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
			NO ACTION					
H-Canyon (limiting solution source term)								
Unpropagated fire in solution vessel	0.594	2.02E-02	2.15	0.355	2.62E+03	1.7E-05 8.6E-04	**3.6E-06** 1.8E-04	2.6E-02 1.3
Inadvertent transfer from a processing vessel to the ground outside the H-Canyon building	1.32	4.00E-04	31.0	1.42	8.27E+03	1.0E-05 2.4E-02	2.8E-07 7.1E-04	1.7E-03 **4.1**
Inadvertent transfer of solution to H-Canyon sump	3.15E-02	8.10E-02	0.114	1.88E-02	1.39E+02	3.7E-06 4.6E-05	7.6E-07 9.4E-06	5.6E-03 7.0E-02
Airborne release of solutions resulting from coil and tube failure in H-Canyon cooling system	2.08E-02	2.55E-03	0.136	6.25E-03	36	1.4E-07 5.4E-05	8.0E-09 3.1E-06	4.6E-05 1.8E-02
Inadvertent nuclear criticality	4.76E+04	1.56E-03	(b)	1.32E-03	(b)	(b) (b)	1.0E-09 6.6E-07	(b) (b)
Severe earthquake	1.17	2.00E-04	27.4	1.26	7.31E+03	4.4E-06 2.2E-02	1.3E-07 6.3E-04	7.3E-04 3.7
			CONVERSION					
H-Canyon (limiting solution source term)								
Same accident analysis as that for the No-Action Alternative								
HB-Line, Phase II (normal processing)								
Severe earthquake	7.00E-04	2.00E-04	1.79E-02	8.82E-04	4.83	1.4E-09 7.2E-06	8.3E-11 4.1E-07	4.8E-07 2.4E-03
Unpropagated fire in gloveboxes	1.60E-03	4.70E-02	6.46E-03	1.07E-03	7.88	1.2E-07 2.6E-06	2.5E-08 5.4E-07	1.9E-04 3.9E-03

Table E-7. (continued).

			Accident consequences			Latent cancer fatalities (LCF)		
Accident	Quantity released (curies)	Frequency (per year)	Uninvolved worker (rem)	MEI[a] (rem)	Offsite population (person-rem)	Uninvolved worker	MEI	Offsite population
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
			INTERIM STORAGE					
Existing Vaults (235-F)								
Rupture storage container (e.g., radiolytic decay)	5.14E-04	2.00E-02	8.62E-04	1.43E-04	1.05	6.9E-09 3.4E-07	1.4E-09 7.2E-08	1.1E-05 5.3E-04
Severe earthquake	1.05E-02	2.00E-04	0.60	7.0E-03	10	4.8E-08 2.4E-04	7.0E-10 3.5E-06	1.0E-06 5.0E-03
Fire	2.0E-5	5.0E-02	6.0E-04	2.0E-05	0.10	1.2E-08 2.4E-07	5.0E-10 1.0E-08	2.5E-06 5.0E-05
High-Level Waste Tanks								
Severe earthquake	(b)	2.00E-04	(b)	3.41E-03	0.26	(b) (b)	3.4E-10 1.7E-06	2.6E-08 1.3E-04
Hydrogen explosion in a tank	(b)	2.00E-05	0.291	1.13E-02	0.43	2.3E-09 1.2E-04	1.1E-10 5.7E-06	4.3E-09 2.2E-04
Waste tank filter fire	(b)	2.5E-02	9.55E-02	3.68E-03	8.5	9.6E-07 3.8E-05	4.6E-08 1.8E-06	1.1E-04 4.3E-03
F-Canyon (without dissolver)								
Airborne release of plutonium solution resulting from coil and tube failure in F-Canyon water cooling tower	17.0	4.00E-02	16.5	0.755	4.42E+03	2.6E-04 6.6E-03	**1.5E-05** 3.8E-04	8.8E-02 2.2
Severe earthquake	64.7	2.00E-04	9.91	0.447	2.64E+03	7.9E-07 4.0E-03	4.5E-08 2.2E-04	2.6E-04 1.3
Fire in a plutonium process vessel	56.2	6.10E-05	10.6	1.75	1.29E+04	2.6E-07 4.3E-03	5.3E-08 8.8E-04	3.9E-04 **6.5**
Inadvertent nuclear criticality	2.40E+05	1.60E-03	(b)	7.43E-03	12.9	(b) (b)	5.9E-09 3.7E-06	1.0E-05 6.5E-03
Inadvertent transfer of plutonium solution from a processing vessel to the ground outside building	24.9	7.40E-05	1.61	7.24E-02	4.30E+02	4.8E-08 6.4E-04	2.7E-09 3.6E-05	1.6E-05 0.22

Table E-7. (continued).

Accident	Quantity released (curies)	Frequency (per year)	Accident consequences: Uninvolved worker (rem)	Accident consequences: MEI[a] (rem)	Accident consequences: Offsite population (person-rem)	Latent cancer fatalities (LCF): Uninvolved worker	Latent cancer fatalities (LCF): MEI	Latent cancer fatalities (LCF): Offsite population
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
ADDITIONAL CONVERSION								
Actinide Packaging and Storage Facility (FB-Line drying)								
Severe earthquake	1.74	2.00E-04	4.54	0.208	1.22E+03	3.6E-07 1.8E-03	2.0E-08 1.0E-04	1.2E-04 0.62
Inadvertent nuclear criticality	(b)	5.26E-05	(b)	2.64E-03	2.93	(b) (b)	6.9E-11 1.3E-06	7.7E-08 1.5E-03
Propagated fire in gloveboxes	3.37E-03	5.26E-03	1.39E-03	2.29E-04	1.69	2.9E-09 5.5E-07	6.1E-10 1.2E-07	4.5E-06 8.4E-04
DWPF-3								
Defense Waste Processing Facility								
Uncontrolled Reaction	(b)	4.50E-02	1.50E-03	1.70E-04	2.50	2.70E-08 6.00E-07	3.83E-09 8.50E-08	5.63E-05 1.25E-03
Melter Spill	(b)	9.30E-03	2.94E-01	3.40E-02	4.90E+02	1.09E-06 1.18E-04	1.58E-07 1.70E-05	2.28E-03 2.45E-01
Earthquake	(b)	5.20E-05[c]	4.04E+03	6.77	7.60E+04	(d) (d)	1.76E-07 3.38E-03	1.98E-03 **3.80E+01**
FB-Line (processing)								
Severe earthquake	4.34	2.00E-04	11.3	0.521	3.06E+03	9.0E-07 4.5E-03	5.2E-08 2.6E-04	3.0E-04 1.5
Inadvertent nuclear criticality in processing solution or solid	(b)	1.40E-04	(b)	2.64E-03	2.93	(b) (b)	1.8E-10 1.3E-06	2.1E-07 1.5E-03
Propagated fire in processing vessels or gloveboxes	0.105	5.26E-03	4.33E-02	7.13E-03	52.7	9.1E-08 1.7E-05	1.9E-08 1.3E-06	1.4E-04 2.6E-02

TC

Table E-7. (continued).

Accident	Quantity released (curies)	Frequency (per year)	Accident consequences: Uninvolved worker (rem)	Accident consequences: MEI[a] (rem)	Accident consequences: Offsite population (person-rem)	Latent cancer fatalities (LCF): Uninvolved worker	Latent cancer fatalities (LCF): MEI	Latent cancer fatalities (LCF): Offsite population
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
F-Canyon (second plutonium cycle contribution)								
Airborne release of plutonium solution resulting from coil and tube failure in F-Canyon water cooling tower	0.218	4.00E-02	0.531	2.44E-02	1.44E+02	8.8E-06 2.2E-04	**4.8E-07** 1.2E-05	2.9E-03 7.2E-02
Severe earthquake	0.365	2.00E-04	3.43	0.158	9.22E+02	2.8E-07 1.4E-03	1.6E-08 7.9E-05	9.2E-05 0.46
Fire in a plutonium process vessel	1.59	6.10E-05	2.27	0.378	2.78E+03	5.5E-08 9.0E-04	1.2E-08 1.9E-04	8.5E-05 1.4
Inadvertent nuclear criticality	2.40E+05	1.60E-03	(b)	7.43E-03	12.9	(b) (b)	5.9E-09 3.7E-06	1.0E-05 6.5E-03
Inadvertent transfer of plutonium solution from a processing vessel to the ground outside building.	9.65E-02	7.40E-05	0.872	4.02E-02	2.35E+02	2.6E-08 3.5E-04	1.5E-09 2.0E-05	8.7E-06 0.12
POST-STABILIZATION STORAGE								
Storage Vault								
Severe earthquake	1.05E-02	2.00E-04	0.60	7.0E-03	10	4.8E-08 2.4E-04	**7.0E-10** 3.5E-06	1.0E-06 **5.0E-03**
Fire	2.0E-05	5.0E-02	6.0E-04	2.0E-05	0.10	1.2E-08 2.4E-07	5.0E-10 1.0E-08	2.5E-06 5.0E-05

TC

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a. MEI = maximally exposed individual.
b. These data were not available.
c. This is the frequency due to the postulated sequence of events; it is based on earthquake frequency of 0.0002 event per year.
d. The number of latent cancer fatalities is not calculated because the dose (4,000 rem) would result in death within a few days.

TC

Table E-8. H-Canyon enriched uranium solutions.

TE

			Accident consequences			Latent cancer fatalities (LCF)		
Accident	Quantity released (curies)	Frequency (per year)	Uninvolved worker (rem)	MEI[a] (rem)	Offsite population (person-rem)	Uninvolved worker	MEI	Offsite population
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
NO ACTION								
H-Canyon (normal solution term)								
Unpropagated fire in solution vessel	2.90E-02	3.56E-02	8.92E-03	5.41E-02	65.8	1.3E-07 3.6E-06	**9.6E-07** 2.7E-05	1.2E-03 3.3E-02
Inadvertent transfer from a processing vessel to the ground outside the H-Canyon building	5.61E-02	4.03E-04	3.11E-02	0.678	1.81E+02	5.0E-09 1.2E-05	1.4E-07 3.4E-04	3.6E-05 9.1E-02
Inadvertent transfer of solution to H-Canyon sump	2.75E-04	8.06E-02	8.43E-05	5.11E-04	0.622	2.7E-09 3.4E-08	2.1E-08 2.6E-07	2.5E-05 3.1E-04
Airborne release of solutions resulting from coil and tube failure in H-Canyon cooling system	19.6	2.55E-03	1.15E-02	0.253	67.6	1.2E-08 4.6E-06	3.2E-07 1.3E-04	8.6E-05 3.4E-02
Inadvertent nuclear criticality	4.76E+04	1.56E-03	(b)	1.32E-03	(b)	(b) (b)	1.0E-09 6.6E-07	(b) (b)
Severe earthquakes	0.149	2.00E-04	2.53E-03	1.15E-04	0.674	2.0E-10 1.0E-06	1.2E-11 5.8E-08	6.7E-08 3.4E-04
H-Outside (UNH tank)								
Transfer error	(b)	1.75E-02	(b)	4.30E-05	0.286	(b) (b)	3.8E-10 2.2E-08	2.5E-06 1.4E-04
Liquid release due to severe earthquake	(b)	2.00E-04	(b)	4.58E-02	2.72E+02	(b) (b)	4.6E-09 2.3E-05	2.8E-05 **0.14**

Table E-8. (continued).

			Accident consequences			Latent cancer fatalities (LCF)		
Accident	Quantity released (curies)	Frequency (per year)	Uninvolved worker (rem)	MEI[a] (rem)	Offsite population (person-rem)	Uninvolved worker	MEI	Offsite population
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
			CONVERSION					
FA-Line (normal operations)								
Eructation (spewing from overpresurization) in vessel during processing	3.40E-05	4.00E-02	1.97E-04	9.04E-06	5.49E-02	3.2E-09	1.8E-10	1.1E-06
						7.9E-08	4.5E-09	2.7E-05
"Red-oil" explosion (i.e., uncontrollable reaction of contaminated organic materials) in the denitrator	2.30E-05	1.40E-04	1.33E-04	6.12E-06	3.71E-02	7.4E-12	4.3E-13	2.6E-09
						5.3E-08	3.1E-09	1.9E-05
Design-basis tornado	2.60	1.00E-06	(b)	2.9E-05	8.0	(b)	1.5E-14	4.0E-09
						(b)	1.5E-08	4.0E-03
Severe earthquake	1.29E-06	2.00E-04	7.47E-06	3.43E-07	2.08E-03	6.0E-13	3.4E-14	2.1E-10
						3.0E-09	1.7E-10	1.0E-06
Uranium Solidification Facility (normal operations)								
Severe earthquake	(b)	2.00E-04	5.90E-02	1.01E-04	0.700	4.7E-09	1.0E-11	7.0E-08
						2.4E-05	5.1E-08	3.5E-04
Uncontrolled chemical reaction during processing in denitrator pot	4.91E-07	4.90E-06	5.25E-07	8.22E-08	6.26E-04	1.0E-15	2.0E-16	1.5E-12
						2.1E-10	4.1E-11	3.1E-07
Inadvertent criticality	(b)	2.27E-04	16.4	1.33E-02	18.8	1.5E-06	1.5E-09	2.1E-06
						6.6E-03	6.7E-06	9.4E-03
H-Canyon (limiting solution source term)								
Unpropagated fire in solution vessel	0.594	2.02E-02	2.15	0.355	2.62E+03	1.7E-05	3.6E-06	2.6E-02
						8.6E-04	1.8E-04	1.3
Inadvertent transfer from a processing vessel to the ground outside the H-Canyon building	1.32	4.00E-04	31.0	1.42	8.27E+03	1.0E-05	2.8E-07	1.7E-03
						2.4E-02	7.1E-04	4.1

Table E-8. (continued).

Accident	Quantity released (curies)	Frequency (per year)	Accident consequences: Uninvolved worker (rem)	Accident consequences: MEI[a] (rem)	Accident consequences: Offsite population (person-rem)	Latent cancer fatalities (LCF): Uninvolved worker	Latent cancer fatalities (LCF): MEI	Latent cancer fatalities (LCF): Offsite population
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
CONVERSION (continued)								
H-Canyon (limiting solution source term) (continued)								
Inadvertent transfer of solution to H-Canyon sump	3.15E-02	8.10E-02	0.114	1.88E-02	1.39E+02	3.7E-06 4.6E-05	7.6E-07 9.4E-06	5.6E-03 7.0E-02
Airborne release of solutions resulting from coil and tube failure in H-Canyon cooling system	2.08E-02	2.55E-03	0.136	6.25E-03	36	1.39E-07 5.44E-05	7.97E-09 3.13E-06	4.59E-05 1.8E-02
Inadvertent nuclear criticality	4.76E+04	1.56E-03	(b)	1.32E-03	(b)	(b) (b)	1.0E-09 6.6E-07	(b) (b)
Severe earthquake	1.17	2.00E-04	27.4	1.26	7.31E+03	4.4E-06 2.2E-02	1.3E-07 6.3E-04	7.3E-04 3.7
INTERIM STORAGE								
High-Level Waste Tanks								
Severe earthquake	(b)	2.00E-04	(b)	3.41E-03	0.26	(b) (b)	3.4E-10 1.7E-06	2.6E-08 1.3E-04
Hydrogen explosion in a tank	(b)	2.00E-05	0.291	1.13E-02	0.43	2.3E-09 1.2E-04	1.1E-10 5.7E-06	4.3E-09 2.2E-04
Waste tank filter fire	(b)	2.5E-02	9.55E-02	3.68E-03	8.5	9.6E-07 3.8E-05	**4.6E-08** 1.8E-06	1.1E-04 **4.3E-03**
ADDITIONAL CONVERSION								
Defense Waste Processing Facility								
Uncontrollable reaction	(b)	4.50E-02	1.50E-03	1.70E-04	2.50	2.70E-08 6.00E-07	3.83E-09 8.50E-08	5.63E-05 1.25E-03
Melter spill	(b)	9.30E-03	2.94E-01	3.40E-02	4.90E+02	1.09E-06 1.18E-04	1.58E-07 1.70E-05	2.28E-03 2.45E-01
Earthquake	(b)	5.20E-05[c]	4.04E+03	6.77	7.60E+04	(d) (d)	**1.76E-07** 3.83E-03	1.98E-03 **3.80E+01**

TC

Table E-8. (continued).

			Accident consequences			Latent cancer fatalities (LCF)		
Accident	Quantity released (curies)	Frequency (per year)	Uninvolved worker (rem)	MEI[a] (rem)	Offsite population (person-rem)	Uninvolved worker	MEI	Offsite population
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
POST-STABILIZATION STORAGE								
Storage Vault (Uranium Storage Facility)								
Severe earthquake	(b)	2.00E-04	5.90E-02	1.01E-04	0.700	4.7E-09	1.0E-11	7.0E-08
						2.4E-05	5.1E-08	3.5E-04
Inadvertent criticality	(b)	2.27E-04	16.4	1.33E-02	18.8	1.5E-06	1.5E-09	2.1E-06
						6.6E-03	6.7E-06	9.4E-03

TC

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a. MEI = Maximally exposed individual.

b. These data were not available.

c. This is the frequency due to the postulated sequence of events, it is based on earthquake frequency of 0.002 event per year.

d. The number of latent cancer fatalities is not calculated because the dose (4,000 rem) would result in death within a few days.

TC

Table E-9. Plutonium and uranium stored in vaults.

TE

TC

Accident	Quantity released (curies)	Frequency (per year)	Accident consequences: Uninvolved worker (rem)	Accident consequences: MEI[a] (rem)	Accident consequences: Offsite population (person-rem)	Latent cancer fatalities (LCF): Uninvolved worker	Latent cancer fatalities (LCF): MEI	Latent cancer fatalities (LCF): Offsite population
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
NO ACTION								
HB Line/235-F (storage)								
Rupture storage container (e.g. radiolytic decay)	5.14E-04	2.00E-02	8.62E-04	1.43E-04	1.05	6.9E-09 3.4E-07	1.4E-09 7.2E-08	1.1E-05 5.3E-04
Severe earthquake	1.05E-02	2.00E-04	0.60	7.0E-03	10	4.8E-08 2.4E-04	7.0E-10 3.5E-06	1.0E-06 5.0E-03
Fire	2.0E-05	5.0E-02	6.0E-04	2.0E-05	0.10	1.2E-08 2.4E-07	5.0E-10 1.0E-08	2.5E-06 5.0E-05
FB-Line (storage)								
Severe earthquake	0.868	2.00E-04	2.27	0.104	6.12E+02	1.8E-07 9.1E-04	**1.0E-08** 5.1E-05	6.1E-05 **0.31**
Inadvertent nuclear criticality in storage and vaults	(b)	8.76E-05	(b)	1.60E-03	1.71	(b) (b)	7.0E-11 8.0E-07	7.5E-08 8.6E-04
CONVERSION								
HB-Line (Phase I with americium contribution)								
Propagated fire in gloveboxes containing plutonium processing vessels	0.615	5.26E-03	2.05	0.338	2.49E+03	4.3E-06 8.2E-04	8.9E-07 1.7E-04	6.5E-03 1.3
Severe earthquake	4.00E-02	2.0E-04	0.910	4.07E-02	2.43E+02	7.3E-08 3.6E-04	4.1E-09 2.0E-05	2.4E-05 0.12

Table E-9. (continued).

Accident	Quantity released (curies)	Frequency (per year)	Accident consequences: Uninvolved worker (rem)	Accident consequences: MEI[a] (rem)	Accident consequences: Offsite population (person-rem)	Latent cancer fatalities (LCF): Uninvolved worker	Latent cancer fatalities (LCF): MEI	Latent cancer fatalities (LCF): Offsite population
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
CONVERSION (continued)								
H-Canyon (limiting solution source term)								
Unpropagated fire in solution vessel	0.594	2.02E-02	2.15	0.355	2.62E+03	1.7E-05 8.6E-04	**3.6E-06** 1.8E-04	2.6E-02 1.3
Inadvertent transfer from a processing vessel to the ground outside the H-Canyon building	1.32	4.00E-04	31.0	1.42	8.27E+03	1.0E-05 2.4E-02	2.8E-07 7.1E-04	1.7E-03 4.1 TC
Inadvertent transfer of solution to H-Canyon sump	3.15E-02	8.10E-02	0.114	1.88E-02	1.39E+02	3.7E-06 4.6E-05	7.6E-07 9.4E-06	5.6E-03 7.0E-02
Airborne release of solutions resulting from coil and tube failure in H-Canyon cooling system	2.08E-02	2.55E-03	0.136	6.25E-03	36	1.4E-07 5.4E-05	8.0E-09 3.1E-06	4.6E-05 1.8E-02
Inadvertent nuclear criticality	4.76E+04	1.56E-03	(b)	1.32E-03	(b)	(b) (b)	1.0E-09 6.6E-07	(b) (b)
Severe earthquake	1.17	2.00E-04	27.4	1.26	7.31E+03	4.4E-06 2.2E-02	1.3E-07 6.3E-04	7.3E-04 3.7
HB-Line Phase II (normal processing)								
Severe earthquake	7.00E-04	2.00E-04	1.79E-02	8.28E-04	4.83	1.4E-09 7.2E-06	8.3E-11 4.1E-07	4.8E-07 2.4E-03
Unpropagated fire in gloveboxes	1.60E-03	4.70E-02	6.46E-03	1.07E-03	7.88	1.2E-07 2.6E-06	2.5E-08 5.4E-07	1.9E-04 3.9E-03
HB-Line Phase I (Pu-238 Recovery)								
Propagated fire	(b)	5.26E-03	0.185	0.100	1.21E+03	3.9E-07 7.4E-05	2.6E-07 5.0E-05	3.2E-03 0.61
Medium energetic event	(b)	3.7E-03	2.18E-02	1.18E-02	1.43E+02	3.2E-08 8.7E-06	2.2E-08 5.9E-06	2.7E-04 7.2E-02
Severe earthquake	(b)	2.0E-04	7.48E-02	9.27E-03	77.4	6.0E-09 3.0E-05	9.3E-10 4.6E-06	7.7E-06 3.9E-02

Table E-9. (continued).

			Accident consequences			Latent cancer fatalities (LCF) (Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
Accident	Quantity released (curies)	Frequency (per year)	Uninvolved worker (rem)	MEI[a] (rem)	Offsite population (person-rem)	Uninvolved worker	MEI	Offsite population
CONVERSION (continued)								
H-Canyon (Frame waste Recovery)								
Severe earthquake	(b)	2.0E-04	1.09	1.1	9.03E+03	8.7E-08 4.4E-04	1.1E-07 5.5E-04	9.0E-04 4.5
Fire	(b)	2.11E-05	0.562	0.303	3.63E+03	4.7E-09 2.3E-04	3.2E-09 1.5E-04	3.8E-05 1.8
Uncontrolled reaction	(b)	7.9E-02	5.86E-02	3.16E-02	31.6	1.9E-06 2.3E-05	1.3E-06 1.6E-05	1.3E-03 1.6E-02
Transfer error to outside	(b)	4.0E-04	1.25	0.672	8.05E+03	2.0E-07 5.0E-04	1.3E-07 3.1E-04	1.6E-03
Coil and tube failure	(b)	1.5E-02	(b)	0.290	7.6E+03	(b) (b)	2.2E-06 1.5E-04	5.7E-02 3.8
HB-Line Phase III (Normal Processing)								
Propagated fire	(b)	(b)	4.9E-02	2.67E-02	3.24E+02	(b) 2.0E-05	(b) 1.3E-05	(b) 0.16
Medium energetic event	(b)	7.0E-04	7.85E-03	4.24E-03	51.4	2.2E-09 3.1E-06	1.5E-09 2.1E-06	1.8E-05 2.6E-02
Severe earthquake	(b)	2.00E-04	2.00E-02	2.48E-03	20.7	1.6E-09 8.0E-06	2.4E-10 1.2E-06	2.1E-06 1.0E-02
Actinide Packaging and Storage Facility (FB-Line drying)								
Severe earthquake	1.74	2.00E-04	4.54	0.208	1.22E+03	3.6E-07 1.8E-03	2.0E-08 1.0E-04	1.2E-04 0.62
Inadvertent nuclear criticality	(b)	5.26E-05	(b)	2.64E-03	2.93	(b) (b)	6.9E-11 1.3E-06	7.7E-08 1.5E-03
Propagated fire in a glovebox	3.37E-03	5.26E-03	1.39E-03	2.29E-04	1.69	2.9E-09 5.5E-07	6.1E-10 1.2E-07	4.5E-06 8.4E-04

Table E-9. (continued).

			Accident consequences			Latent cancer fatalities (LCF)		
Accident	Quantity released (curies)	Frequency (per year)	Uninvolved worker (rem)	MEI[a] (rem)	Offsite population (person-rem)	Uninvolved worker	MEI	Offsite population
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
INTERIM STORAGE								
Existing storage								
Rupture storage container (e.g., radiolytic decay)	5.14E-04	2.00E-02	8.62E-04	1.43E-04	1.05	6.9E-09 3.4E-07	1.4E-09 7.2E-08	1.1E-05 5.3E-04
Severe earthquake	1.05E-02	2.00E-04	0.60	7.0E-03	10	4.8E-08 2.4E-04	7.0E-10 3.5E-06	1.0E-06 **5.0E-03**
Fire	2.0E-05	5.0E-02	6.0E-04	2.0E-05	0.10	1.2E-08 2.4E-07	5.0E-10 1.0E-08	2.5E-06 5.0E-05
High-Level Waste Tanks								
Severe earthquake	(b)	2.00E-04	(b)	3.41E-03	0.26	(b) (b)	3.4E-10 1.7E-06	2.6E-08 1.3E-04
Hydrogen explosion in a tank	(b)	2.00E-05	0.291	1.13E-02	0.43	2.3E-09 1.2E-04	1.1E-10 5.7E-06	4.3E-09 2.2E-04
Waste tank filter fire	(b)	2.5E-02	9.55E-02	3.68E-03	8.5	9.6E-07 3.8E-05	**4.6E-08** 1.8E-06	1.1E-04 4.3E-03
ADDITIONAL CONVERSION								
FB-Line (processing)								
Severe earthquake	4.34	2.00E-04	11.3	0.521	3.06E+03	9.0E-07 4.5E-03	5.2E-08 2.6E-04	3.0E-04 1.5
Inadvertent nuclear criticality in processing solution or solid	(b)	1.40E-04	(b)	2.64E-03	2.93	(b) (b)	1.8E-10 1.3E-06	2.1E-07 1.5E-03
Propagated fire in processing vessels or gloveboxes	0.105	5.26E-03	4.33E-02	7.13E-03	52.7	9.1E-08 1.7E-05	1.9E-08 1.3E-06	1.4E-04 2.6E-02

Table E-9. (continued).

Accident	Quantity released (curies)	Frequency (per year)	Accident consequences: Uninvolved worker (rem)	Accident consequences: MEI[a] (rem)	Accident consequences: Offsite population (person-rem)	Latent cancer fatalities (LCF): Uninvolved worker	Latent cancer fatalities (LCF): MEI	Latent cancer fatalities (LCF): Offsite population
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
ADDITIONAL CONVERSION (continued)								
Actinide Packaging and Storage Facility (FB-Line drying)								
Severe earthquake	1.74	2.00E-04	4.54	0.208	1.22E+03	3.6E-07 1.8E-03	2.0E-08 1.0E-04	1.2E-04 0.62
Inadvertent nuclear criticality	(b)	5.26E-05	(b)	2.64E-03	2.93	(b) (b)	6.9E-11 1.3E-06	7.7E-08 1.5E-03
Propagated fire in gloveboxes	3.37E-03	5.26E-03	1.39E-03	2.29E-04	1.69	2.9E-09 5.5E-07	6.1E-10 1.2E-07	4.5E-06 8.4E-04
Defense Waste Processing Facility								
Uncontrolled Reaction	(b)	4.50E-02	1.50E-03	1.70E-04	2.50	2.70E-08 6.00E-07	3.83E-09 8.50E-08	5.63E-05 1.25E-03
Melter Spill	(b)	9.30E-03	2.94E-01	3.40E-02	4.90E+02	1.09E-06 1.18E-04	1.58E-07 1.70E-05	2.28E-03 2.45E-01
Earthquake	(b)	5.20E-05[c]	4.04E+03	6.77	7.60E+04	(d) (d)	1.76E-07 3.38E-03	1.98E-03 **3.80E+01**
F-Canyon (second plutonium cycle contribution)								
Airborne release of plutonium solution resulting from coil and tube failure in F-Canyon water cooling tower	0.218	4.00E-02	0.531	2.44E-02	1.44E+02	8.8E-06 2.2E-04	**4.84E-07** 1.2E-05	2.9E-03 7.2E-02
Severe earthquake	0.365	2.00E-04	3.43	0.158	9.22E+02	2.8E-07 1.4E-03	1.6E-08 7.9E-05	9.2E-05 0.46
Fire in a plutonium process vessel	1.59	6.10E-05	2.27	0.378	2.78E+03	5.5E-08 9.0E-04	1.2E-08 1.9E-04	8.5E-05 1.4
Inadvertent nuclear criticality	2.40E+05	1.60E-03	(b)	7.43E-03	12.9	(b) (b)	5.9E-09 3.7E-06	1.0E-05 6.5E-03
Inadvertent transfer of plutonium solution from a processing vessel to the ground outside building	9.65E-02	7.40E-05	0.872	4.02E-02	2.35E+02	2.6E-08 3.5E-04	1.5E-09 2.0E-05	8.7E-06 0.12

Table E-9. (continued).

			Accident consequences			Latent cancer fatalities (LCF)		
Accident	Quantity released (curies)	Frequency (per year)	Uninvolved worker (rem)	MEI[a] (rem)	Offsite population (person-rem)	Uninvolved worker	MEI	Offsite population
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
POST-STABILIZATION STORAGE								
Storage Vault								
Severe earthquake	1.05E-02	2.00E-04	0.60	7.0E-03	10	4.8E-08 2.4E-04	**7.0E-10** 3.5E-06	1.0E-06 **5.0E-03**
Fire	2.0E-05	5.0E-02	6.0E-04	2.0E-05	0.10	1.2E-08 2.4E-07	5.0E-10 1.0E-08	2.5E-06 5.0E-05
Inadvertent criticality in storage	(b)	8.76E-05	(b)	1.60E-03	1.71	(b) (b)	7.0E-11 8.0E-07	7.5E-08 8.6E-04

TC

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a. MEI = maximally exposed individual.
b. These data were not available.
c. This is the frequency due to the postulated sequence of events; it is based on earthquake frequency of 2.0E-04 events per year.
d. The number of latent cancer fatalities is not calculated because the dose (4,000 rem) would result in death within a few days.

TC

Table E-10. Mark-31 targets.

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			Accident consequences			Latent cancer fatalities (LCF)		
Accident	Quantity released (curies)	Frequency (per year)	Uninvolved worker (rem)	MEI[a] (rem)	Offsite population (person-rem)	Uninvolved worker	MEI	Offsite population
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
			NO ACTION					
L-Reactor Basin (storage)								
Inadvertent draindown of half the basin water to the Savannah River	2.57E+03	1.08E-02	(b)	9.12E-04	0.678	(b) (b)	**4.9E-09** 4.6E-07	3.7E-06 3.4E-04
Severe earthquake	4.27E+05	2.00E-04	7.64E-04	5.36E-03	17.7	6.1E-09 3.1E-05	5.4E-10 2.7E-06	1.8E-06 **8.9E-03**
Inadvertent overflow of 37,850 liters[c] of basin water through sewer system to Savannah River	15.1	1.56E-02	(b)	5.37E-06	3.99E-03	(b) (b)	4.2E-11 2.7E-09	3.1E-08 2.0E-06
			CONVERSION					
F-Canyon (full operations)								
Airborne release of plutonium solution resulting from coil and tube failure in F-Canyon water cooling tower	17.0	4.00E-02	16.5	0.755	4.42E+03	2.6E-04 6.6E-03	**1.5E-05** 3.8E-04	**8.8E-02** 2.2
Severe earthquake	73.0	2.00E-04	10.5	0.474	2.80E+03	8.4E-07 4.2E-03	4.7E-08 2.4E-04	2.8E-04 1.4
Fire in a plutonium process vessel	56.2	6.10E-05	10.6	1.75	1.29E+04	2.6E-07 4.2E-03	5.3E-08 8.8E-04	3.9E-04 **6.5**
Ruthenium volatilization	30.0	5.30E-02	0.105	1.77E-02	1.29E+02	2.2E-06 4.2E-05	4.7E-07 8.9E-06	3.4E-03 6.5E-02
Inadvertent nuclear criticality	2.40E+05	1.60E-03	(b)	7.43E-03	12.9	(b) (b)	5.9E-09 3.7E-06	1.0E-05 6.5E-03
Inadvertent transfer of plutonium solution from a processing vessel to the ground outside building	24.9	1.10E-04	1.61	7.24E-02	4.30E+02	7.1E-08 6.4E-04	4.0E-09 3.6E-05	2.4E-05 0.22

Table E-10. (continued).

			Accident consequences			Latent cancer fatalities (LCF)		
Accident	Quantity released (curies)	Frequency (per year)	Uninvolved worker (rem)	MEI[a] (rem)	Offsite population (person-rem)	Uninvolved worker	MEI	Offsite population
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
			CONVERSION (continued)					
FA-Line (normal operations)								
Eructation (spewing from overpressurization) in vessel during processing	3.40E-05	4.00E-02	1.97E-04	9.04E-06	5.49E-02	3.2E-09 7.9E-08	1.8E-10 4.5E-09	1.1E-06 2.7E-05
"Red oil" explosion (i.e., uncontrollable reaction of contaminated organic materials) in denitrator	2.30E-05	1.40E-04	1.33E-04	6.12E-06	3.71E-02	7.4E-12 5.3E-08	4.3E-13 3.1E-09	2.6E-09 1.9E-05
Design-basis tornado	2.60	1.00E-06	(b)	2.9E-05	8.0	(b) (b)	1.5E-14 1.5E-08	4.0E-09 4.0E-03
Severe earthquake	1.29E-06	2.00E-04	7.47E-06	3.43E-07	2.08E-03	6.0E-13 3.0E-09	3.4E-14 1.7E-10	2.1E-10 1.0E-06
FB-Line (processing)								
Severe earthquake	4.34	2.00E-04	11.3	0.521	3.06E+03	9.0E-07 4.5E-03	5.2E-08 2.6E-04	3.0E-04 1.5
Inadvertent nuclear criticality in processing solution or solid	(b)	1.40E-04	(b)	2.64E-03	2.93	(b) (b)	1.8E-10 1.3E-06	2.1E-07 1.5E-03
Propagated fire in processing vessels or gloveboxes	0.105	5.26E-03	4.33E-02	7.13E-03	52.7	9.1E-08 1.7E-05	1.9E-08 3.6E-06	1.4E-04 2.6E-02
			INTERIM STORAGE					
Existing Vaults (235-F)								
Rupture storage container (e.g., radiolytic decay)	5.14E-04	2.00E-02	8.62E-04	1.43E-04	1.05	6.9E-09 3.4E-07	1.4E-09 7.2E-08	1.1E-05 5.3E-04
Severe earthquake	1.05E-02	2.00E-04	0.60	7.0E-03	10	4.8E-08 2.4E-04	7.0E-10 3.5E-06	1.0E-06 **5.0E-03**
Fire	2.0E-05	5.0E-02	6.0E-04	2.0E-05	0.10	1.2E-08 2.4E-07	5.0E-10 1.0E-08	2.5E-06 5.0E-05

Table E-10. (continued).

Accident	Quantity released (curies)	Frequency (per year)	Accident consequences			Latent cancer fatalities (LCF) (Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
			Uninvolved worker (rem)	MEI[a] (rem)	Offsite population (person-rem)	Uninvolved worker	MEI	Offsite population
INTERIM STORAGE (continued)								
High-Level Waste Tanks								
Severe earthquake	(b)	2.00E-04	(b)	3.41E-03	0.26	(b) (b)	3.4E-10 1.7E-06	2.6E-08 1.3E-04
Hydrogen explosion in a tank	(b)	2.00E-05	0.291	1.13E-02	0.43	2.3E-09 1.2E-04	1.1E-10 5.7E-06	4.3E-09 2.2E-04
Waste tank filter fire	(b)	2.5E-02	9.55E-02	3.68E-03	8.5	9.6E-07 3.8E-05	**4.6E-08** 1.8E-06	1.1E-04 4.3E-03
ADDITIONAL CONVERSION								
Actinide Packaging and Storage Facility (FB-Line drying)								
Severe earthquake	1.74	2.00E-04	4.54	0.208	1.22E+03	3.6E-07 1.8E-03	2.0E-08 1.0E-04	1.2E-04 0.62
Inadvertent nuclear criticality	(b)	5.26E-05	(b)	2.64E-03	2.93	(b) (b)	6.9E-11 1.3E-06	7.7E-08 1.5E-03
Propagated fire in a glovebox	3.37E-03	5.26E-03	1.39E-03	2.29E-04	1.69	2.9E-09 5.5E-07	6.1E-10 1.2E-07	4.5E-06 8.4E-04
Defense Waste Processing Facility								
Uncontrolled reaction	(b)	4.50E-02	1.50E-03	1.70E-04	2.50	2.70E-08 6.00E-07	3.83E-09 8.50E-08	5.63E-05 1.25E-03
Melter spill	(b)	9.30E-03	2.94E-01	3.40E-02	4.90E+02	1.09E-06 1.18E-04	1.58E-07 1.7.0E-05	2.28E-03 2.45E-01
Earthquake	(b)	5.20E-05[d]	4.04E+03	6.77	7.60E+04	(e) (e)	1.76E-07 3.38E-03	1.98E-03 **3.80E+01**

Table E-10. (continued).

			Accident consequences			Latent cancer fatalities (LCF)		
Accident	Quantity released (curies)	Frequency (per year)	Uninvolved worker (rem)	MEI[a] (rem)	Offsite population (person-rem)	Uninvolved worker	MEI	Offsite population
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
ADDITIONAL CONVERSION (continued)								
F-Canyon (second plutonium cycle contribution)								
Airborne release of plutonium solution resulting from coil and tube failure in F-Canyon water cooling tower	0.218	4.00E-02	0.531	2.44E-02	1.44E+02	8.8E-06	**4.8E-07**	2.9E-03
						2.2E-04	1.2E-05	7.2E-02
Severe earthquake	0.365	2.00E-04	3.43	0.158	9.22E+02	2.8E-07	1.6E-08	9.2E-05
						1.4E-03	7.9E-05	0.6
Fire in a plutonium process vessel	1.59	6.10E-05	2.27	0.378	2.78E+03	5.5E-08	1.2E-08	8.5E-05
						9.0E-04	1.9E-04	1.4
Inadvertent nuclear criticality	2.40E+05	1.60E-03	(b)	7.43E-03	12.9	(b)	5.9E-09	1.0E-05
						(b)	3.7E-06	6.5E-03
Inadvertent transfer of plutonium solution from a processing vessel to the ground outside building.	9.65E-02	7.40E-05	0.872	4.02E-02	2.35E+02	2.6E-08	1.5E-09	8.7E-06
						3.5E-04	2.0E-05	0.12
POST-STABILIZATION STORAGE								
Storage Vault								
Severe earthquake	1.05E-02	2.00E-04	0.60	7.0E-03	10	4.8E-08	**7.0E-10**	1.0E-06
						2.4E-04	3.5E-06	**5.0E-03**
Fire	2.0E-05	5.0E-02	6.0E-04	2.0E-05	0.10	1.2E-08	5.0E-10	2.5E-06
						2.4E-07	1.0E-08	5.0E-05

Table E-10. (continued).

Accident	Quantity released (curies)	Frequency (per year)	Accident consequences Uninvolved worker (rem)	MEI[a] (rem)	Offsite population (person-rem)	Latent cancer fatalities (LCF) Uninvolved worker	MEI	Offsite population
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
POST-STABILIZATION STORAGE (continued)								
Dry Storage Facility								
Material release (e.g., assembly breach)	(b)	1.4E-03	(b)	2.1E-06	6.9E-03	(b)	1.5E-12	4.8E-09
						(b)	1.1E-09	3.5E-06

TC

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a. MEI = Maximally exposed individual.

b. These data were not available.

c. To convert liters to gallons, multiply by 0.26418.

d. This is the frequency due to the postulated sequence of events; it is based on earthquake frequency of 0.0002 event per year.

e. The number of latent cancer fatalities is not calculated because the dose (4,000 rem) would result in death within a few days.

TC

Table E-11. Mark-16 and -22 fuel.

			Accident consequences			Latent cancer fatalities (LCF)		
Accident	Quantity released (curies)	Frequency (per year)	Uninvolved worker (rem)	MEI[a] (rem)	Offsite population (person-rem)	Uninvolved worker (Point estimate of increased risk per year) (Increased risk of LCF per occurrence)	MEI	Offsite population
				NO ACTION				TC
Reactor Basins (storage)								
Inadvertent draindown of half the basin water to the Savannah River	2.57E+03	1.08E-02	(b)	9.12E-04	0.678	(b) (b)	**4.9E-09** 4.6E-07	3.7E-06 3.4E-04
Severe earthquake	4.27E+05	2.00E-04	7.64E-02	5.36E-03	17.7	6.1E-09 3.1E-05	5.4E-10 2.7E-06	1.8E-06 **8.9E-03**
Inadvertent overflow 37,850 liters[c] of basin water through sewer system to Savannah River	15.1	1.56E-02	(b)	5.37E-06	3.99E-03	(b) (b)	4.2E-11 2.7E-09	3.1E-08 2.0E-06
				CONVERSION				TC
F/H-Canyon (limiting solution source term)								
Unpropagated fire in solution vessel	0.594	2.02E-02	2.15	0.355	2.62E+03	1.7E-05 8.6E-04	**3.6E-06** 1.8E-04	2.6E-02 1.3
Inadvertent transfer from a processing vessel to the ground outside the H-Canyon building	1.32	4.00E-04	31.0	1.42	8.27E+03	1.0E-05 2.4E-02	2.8E-07 7.1E-04	1.7E-03 **4.1**
Inadvertent transfer of solution to H-Canyon sump	3.15E-02	8.10E-02	0.114	1.88E-02	1.39E+02	3.7E-06 4.6E-05	7.6E-07 9.4E-06	5.6E-03 7.0E-02
Airborne release of solutions resulting from coil and tube failure in H-Canyon cooling system	2.08E-02	2.55E-03	0.136	6.25E-03	36	1.4E-07 5.4E-05	8.0E-09 3.1E-06	4.6E-05 1.8E-02
Inadvertent nuclear criticality	4.76E+04	1.56E-03	(b)	1.32E-03	(b)	(b) (b)	1.0E-09 6.6E-07	(b) (b)
Severe earthquake	1.17	2.00E-04	27.4	1.26	7.31E+03	4.4E-06 2.2E-02	1.3E-07 6.3E-04	7.3E-04 3.7

Table E-11. (continued).

			Accident consequences			Latent cancer fatalities (LCF)		
Accident	Quantity released (curies)	Frequency (per year)	Uninvolved worker (rem)	MEI[a] (rem)	Offsite population (person-rem)	Uninvolved worker	MEI	Offsite population
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
			CONVERSION (continued)					
H-Outside (UNH tank)								
Transfer error	(b)	1.75E-02	(b)	4.30E-05	0.286	(b) (b)	3.8E-10 2.2E-08	2.5E-06 1.4E-04
Liquid release due to severe earthquake	(b)	2.00E-04	(b)	4.58E-02	2.72E+02	(b) (b)	4.6E-09 2.3E-05	2.8E-05 0.14
FA-Line (normal operations)								
Eructation (spewing from overpressurization) in vessel during processing	3.40E-05	4.00E-02	1.97E-04	9.04E-06	5.49E-02	3.2E-09 7.9E-08	1.8E-10 4.5E-09	1.1E-06 2.7E-05
"Red-oil" explosion (i.e., uncontrollable reaction of contaminated organic materials) in the denitrator	2.30E-05	1.40E-04	1.33E-04	6.12E-06	3.71E-02	7.4E-12 5.3E-08	4.3E-13 3.1E-09	2.6E-09 1.9E-05
Design-basis tornado	2.60	1.00E-06	(b)	2.9E-05	8.0	(b) (b)	1.5E-14 1.5E-08	4.0E-09 4.0E-03
Severe earthquake	1.29E-06	2.00E-04	7.47E-06	3.43E-07	2.08E-03	6.0E-13 3.0E-09	3.4E-14 1.7E-10	2.1E-10 1.0E-06
Uranium Solidification Facility (normal operations)								
Severe earthquake	(b)	2.00E-04	5.90E-02	1.01E-04	0.700	4.7E-09 2.4E-05	1.0E-11 5.1E-08	7.0E-08 3.5E-04
Uncontrolled chemical reaction during processing in denitrator pot	4.91E-07	4.90E-06	5.25E-07	8.22E-08	6.26E-04	1.0E-15 2.1E-10	2.0E-16 4.1E-11	1.5E-12 3.1E-07
Inadvertent criticality	(b)	2.27E-04	16.4	1.33E-02	18.8	1.5E-06 6.6E-03	1.5E-09 6.7E-06	2.1E-06 9.4E-03

Table E-11. (continued).

Accident	Quantity released (curies)	Frequency (per year)	Accident consequences: Uninvolved worker (rem)	Accident consequences: MEI[a] (rem)	Accident consequences: Offsite population (person-rem)	Latent cancer fatalities (LCF): Uninvolved worker	Latent cancer fatalities (LCF): MEI	Latent cancer fatalities (LCF): Offsite population
						(Point estimate of increased risk per year)		
						(Increased risk of LCF per occurrence)		
INTERIM STORAGE								
High-Level Waste Tanks								
Severe earthquake	(b)	2.00E-04	(b)	3.41E-03	0.26	(b)	3.4E-10	2.6E-08
						(b)	1.7E-06	1.3E-04
Hydrogen explosion in a tank	(b)	2.00E-05	0.291	1.13E-02	0.43	2.3E-09	1.1E-10	4.3E-09
						1.2E-04	5.7E-06	2.2E-04
Waste tank filter fire	(b)	2.5E-02	9.55E-02	3.68E-03	8.5	9.6E-07	**4.6E-08**	1.1E-04
						3.8E-05	1.8E-06	4.3E-03
ADDITIONAL CONVERSION								
Defense Waste Processing Facility								
Uncontrolled reaction	(b)	4.50E-02	1.50E-03	1.70E-04	2.50	2.70E-08	3.83E-09	5.63E-05
						6.00E-07	8.50E-08	1.25E-03
Melter spill	(b)	9.30E-03	2.94E-01	3.40E-02	4.90E+02	1.09E-06	1.58E-07	2.28E-03
						1.18E-04	1.70E-05	2.45E-01
Earthquake	(b)	5.20E-05[d]	4.04E+03	6.77	7.60E+04	(e)	1.76E-07	1.98E-03
						(e)	3.38E-03	**3.80E+01**
POST-STABILIZATION STORAGE								
Dry Storage Facility								
Material release (e.g., assembly breach)	(b)	1.4E-03	(b)	2.1E-06	6.9E-03	(b)	1.5E-12	4.8E-09
						(b)	1.1E-09	3.5E-06

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a. MEI = maximally exposed individual.
b. These data were not available.
c. To convert liters to gallons, multiply by 0.26418.
d. This is the frequency due to the postulated sequence of events; it is based on earthquake frequency of 0.0002 event per year.
e. The number of latent cancer fatalities is not calculated because the dose (4,000 rem) would result in death within a few days.

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Table E-12. Taiwan Research Reactor and Experimental Breeder Reactor-II fuel.

Accident	Quantity released (curies)	Frequency (per year)	Accident consequences: Uninvolved worker (rem)	Accident consequences: MEI[a] (rem)	Accident consequences: Offsite population (person-rem)	Latent cancer fatalities (LCF): Uninvolved worker	Latent cancer fatalities (LCF): MEI	Latent cancer fatalities (LCF): Offsite population
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
NO ACTION								
Receiving Basin for Offsite Fuels								
Natural phenomenon induced nuclear criticality	(b)	5.20E-04	1.16	3.96E-03	(b)	2.41E-07 4.64E-04	1.03E-09 1.98E-06	(b) (b)
Criticality with 200 fuel ruptures[c]	(b)	3.17E-03	1.17	4.11E-03	(b)	1.48E-06 4.68E-04	6.51E-09 2.06E-06	(b) (b)
CONVERSION								
F-Canyon (full operations)								
Airborne release of plutonium solution resulting from coil and tube failure in F-Canyon water cooling tower	17.0	4.00E-02	16.5	0.755	4.42E+03	2.6E-04 6.6E-03	**1.5E-05** 3.8E-04	**8.8E-02** 2.2
Severe earthquake	73.0	2.00E-04	10.5	0.474	2.80E+03	8.4E-07 4.2E-03	4.7E-08 2.4E-04	2.8E-04 1.4
Fire in a plutonium process vessel	56.2	6.10E-05	10.6	1.75	1.29E+04	2.6E-07 4.2E-03	5.3E-08 8.8E-04	3.9E-04 **6.5**
Ruthenium volatilization	30.0	5.30E-02	0.105	1.77E-02	1.29E+02	2.2E-06 4.2E-05	4.7E-07 8.9E-06	3.4E-03 6.5E-02
Inadvertent nuclear criticality	2.40E+05	1.60E-03	(b)	7.43E-03	12.9	(b) (b)	5.9E-09 3.7E-06	1.0E-05 6.5E-03
Inadvertent transfer of plutonium solution from a processing vessel to the ground outside building	24.9	1.10E-04	1.61	7.24E-02	4.30E+02	7.1E-08 6.4E-04	4.0E-09 3.6E-05	2.4E-05 0.22

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Table E-12. (continued).

			Accident consequences			Latent cancer fatalities (LCF)		
Accident	Quantity released (curies)	Frequency (per year)	Uninvolved worker (rem)	MEI[a] (rem)	Offsite population (person-rem)	Uninvolved worker	MEI	Offsite population
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
CONVERSION (continued)								
FA-Line (normal operations)								
Eructation (spewing from overpressurization) in vessel during processing	3.40E-05	4.00E-02	1.97E-04	9.04E-06	5.49E-02	3.2E-09 7.9E-08	1.8E-10 4.5E-09	1.1E-06 2.7E-05
"Red oil" explosion (i.e., uncontrollable reaction of contaminated organic materials) in denitrator	2.30E-05	1.40E-04	1.33E-04	6.12E-06	3.71E-02	7.4E-12 5.3E-08	4.3E-13 3.1E-09	2.6E-09 1.9E-05
Design-basis tornado	2.60	1.00E-06	(b)	2.9E-05	8.0	(b) (b)	1.5E-14 1.5E-08	4.0E-09 4.0E-03
Severe earthquake[d]	1.29E-06	2.00E-04	7.47E-06	3.43E-07	2.08E-03	6.0E-13 3.0E-09	3.4E-14 1.7E-10	2.1E-10 1.0E-06
FB-Line (processing)								
Severe earthquake	4.34	2.00E-04	11.3	0.521	3.06E+03	9.0E-07 4.5E-03	5.2E-08 2.6E-04	3.0E-04 1.5
Inadvertent nuclear criticality in processing solution or solid	(b)	1.40E-04	(b)	2.64E-03	2.93	(b) (b)	1.8E-10 1.3E-06	2.1E-07 1.5E-03
Propagated fire in processing vessels or gloveboxes	0.105	5.26E-03	4.33E-02	7.13E-03	52.7	9.1E-08 1.7E-05	1.9E-08 3.6E-06	1.4E-04 2.6E-02
INTERIM STORAGE								
Existing Vaults (235-F)								
Rupture storage container (e.g., radiolytic decay)	5.14E-04	2.00E-02	8.62E-04	1.43E-04	1.05	6.9E-09 3.4E-07	1.4E-09 7.2E-08	1.1E-05 5.3E-04
Severe earthquake	1.05E-02	2.00E-04	0.60	7.0E-03	10	4.8E-08 2.4E-04	7.0E-10 3.5E-06	1.0E-06 **5.0E-03**
Fire	2.0E-05	5.0E-02	6.0E-04	2.0E-05	0.10	1.2E-08 2.4E-07	5.0E-10 1.0E-08	2.5E-06 5.0E-05

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Table E-12. (continued).

			Accident consequences			Latent cancer fatalities (LCF)		
Accident	Quantity released (curies)	Frequency (per year)	Uninvolved worker (rem)	MEI[a] (rem)	Offsite population (person-rem)	Uninvolved worker	MEI	Offsite population
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
INTERIM STORAGE (continued)								
High-Level Waste Tanks								
Severe earthquake	(b)	2.00E-04	(b)	3.41E-03	0.26	(b) (b)	3.4E-10 1.7E-06	2.6E-08 1.3E-04
Hydrogen explosion in a tank	(b)	2.00E-05	0.291	1.13E-02	0.43	2.3E-09 1.2E-04	1.1E-10 5.7E-06	4.3E-09 2.2E-04
Waste tank filter fire	(b)	2.5E-02	9.55E-02	3.68E-03	8.5	9.6E-07 3.8E-05	**4.6E-08** 1.8E-06	1.1E-04 4.3E-03
ADDITIONAL CONVERSION								
Actinide Packaging and Storage Facility (FB-Line drying)								
Severe earthquake	1.74	2.00E-04	4.54	0.208	1.22E+03	3.6E-07 1.8E-03	2.0E-08 1.0E-04	1.2E-04 0.62
Inadvertent nuclear criticality	(b)	5.26E-05	(b)	2.64E-03	2.93	(b) (b)	6.9E-11 1.3E-06	7.7E-08 1.5E-03
Propagated fire in a glovebox	3.37E-03	5.26E-03	1.39E-03	2.29E-04	1.69	2.9E-09 5.5E-07	6.1E-10 1.2E-07	4.5E-06 8.4E-04
Defense Waste Processing Facility								
Uncontrolled reaction	(b)	4.50E-02	1.50E-03	1.70E-04	2.50	2.70E-08 6.00E-07	3.83E-09 8.50E-08	5.63E-05 1.25E-03
Melter spill	(b)	9.30E-03	2.94E-01	3.40E-02	4.90E+02	1.09E-06 1.18E-04	1.58E-07 1.70E-05	2.28E-03 2.45E-01
Earthquake	(b)	5.20E-05[d]	4.04E+03	6.77	7.60E+04	(e) (e)	1.76E-07 3.38E-03	1.98E-03 **3.80E+01**

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Table E-12. (continued).

			Accident consequences			Latent cancer fatalities (LCF)		
Accident	Quantity released (curies)	Frequency (per year)	Uninvolved worker (rem)	MEI[a] (rem)	Offsite population (person-rem)	Uninvolved worker	MEI	Offsite population
						(Point estimate of increased risk per year) (Increased risk of LCF per occurrence)		
ADDITIONAL CONVERSION (continued)								
F-Canyon (second plutonium cycle contribution)								
Airborne release of plutonium solution resulting from coil and tube failure in F-Canyon water cooling tower	0.218	4.00E-02	0.531	2.44E-02	1.44E+02	8.8E-06 2.2E-04	**4.8E-07** 1.2E-05	2.9E-03 7.2E-02
Severe earthquake	0.365	2.00E-04	3.43	0.158	9.22E+02	2.8E-07 1.4E-03	1.6E-08 7.9E-05	9.2E-05 0.6
Fire in a plutonium process vessel	1.59	6.10E-05	2.27	0.378	2.78E+03	5.5E-08 9.0E-04	1.2E-08 1.9E-04	8.5E-05 1.4
Inadvertent nuclear criticality	2.40E+05	1.60E-03	(b)	7.43E-03	12.9	(b) (b)	5.9E-09 3.7E-06	1.0E-05 6.5E-03
Inadvertent transfer of plutonium solution from a processing vessel to the ground outside building.	9.65E-02	7.40E-05	0.872	4.02E-02	2.35E+02	2.6E-08 3.5E-04	1.5E-09 2.0E-05	8.7E-06 0.12
POST-STABILIZATION STORAGE								
Storage Vault								
Severe earthquake	1.05E-02	2.00E-04	0.60	7.0E-03	10	4.8E-08 2.4E-04	**7.0E-10** 3.5E-06	1.0E-06 **5.0E-03**
Fire	2.0E-05	5.0E-02	6.0E-04	2.0E-05	0.10	1.2E-08 2.4E-07	5.0E-10 1.0E-08	2.5E-06 5.0E-05
Dry Storage Facility								
Material release (e.g., assembly breach)	(b)	1.4E-03	(b)	2.1E-06	6.9E-03	(b) (b)	1.5E-12 1.1E-09	4.8E-09 3.5E-06

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a. MEI = Maximally exposed individual.
b. These data were not available.
c. The Taiwanese Research Reactor fuel and the Experimental Breeder Reactor fuel occupy approximately 200 spaces or slots in the Receiving Basin for Offsite Fuel. DOE assumed that all would rupture after a criticality, whether previously considered "failed" or not. The dose estimates are based on a bounding Reference Fuel Assembly created for analytical purposes that contains all isotopes found in any fuel, and are, therefore, conservative.
d. This is the frequency due to the postulated sequence of events; it is based on earthquake frequency of 0.0002 event per year.
e. The number of latent cancer fatalities is not calculated because the dose (4,000 rem) would result in death within a few days.

would be applicable for an alternative during a specific phase. However, because the canyons and | TC
their support facilities are similar, conversion activities could occur in either area. As stated above, the
tables list in bold type the maximum point estimate of risk for the maximally exposed offsite
individual and the highest consequence to the population for each phase. Because an alternative
might not involve every facility listed in each phase, these maximum values would not necessarily
apply to all alternatives. For example, the highest point estimate of risk for the conversion phase of
the H-Canyon uranium solutions (0.0000036 latent cancer fatality per year) would occur for | TC
H-Canyon. However, the Low Enriched Uranium Alternative for this material would use FA-Line for
the processing phase; therefore, the maximum point estimate of risk for this alternative during
processing would be 0.00000000018 latent cancer fatality per year. As noted above, the accident | TC
consequences have been tailored to the extent possible to reflect consequences attributable to the
specific material.

Section E.8 includes a glossary of accident descriptions. These descriptions describe the events listed in the tables. The tables use titles that indicate the facility mode as used throughout the tables [e.g., "F-Canyon (without dissolver)]." This entry means the action of dissolving would not be part of the management alternative for this material; the safety analysis report data for this mode or condition has not been used.

E.3.2 IMPACTS TO FACILITY WORKERS FROM POSTULATED FACILITY ACCIDENTS

E.3.2.1 F-Canyon and H-Canyon

No fatalities to involved or "close-in" workers from the accident scenarios postulated under current or full operations in the F- or H-Canyon are a likely result of exposure to radiation. Releases from most accidents would be contained in the processing area and filtered through the canyon ventilation system. Because the ventilation system flows from areas of lowest to highest radioactivity, and because releases flow through an exhaust stack after passing through a filtration system, the doses received by workers from these accidents are not likely to be substantially larger than those received during routine operations. For postulated accidents in which the release was not likely to be maintained within the ventilation system (i.e., airborne releases from the ground level or liquid releases), involved worker exposures would be unlikely to result in adverse health effects. For an inadvertent nuclear criticality in the processing vessels, the doses to involved workers would likely be minimized due to the shielding between the vessels and the locations a worker could occupy.

E.3.2.2 FB-Line Facility

With the exception of an inadvertent nuclear criticality during processing, no fatalities to involved workers from the accident scenarios postulated under current or full operations in the FB-Line would be likely as a result of exposure to radiation (see Section E.7). Current operations primarily involve storage activities in the FB-Line vaults. Because access to storage areas in the FB-Line is limited, only a small number of individuals could receive impacts from an accidental release of material or an inadvertent nuclear criticality in a storage vault. Under full operations, potential accidents resulting from processing, such as a fire or uncontrolled chemical reaction, would not result in substantial exposures because most work would occur inside gloveboxes. Based on historic accident information, exposures to involved workers would be within limits established for routine operations if the implementation of emergency response actions occurred. Of the approximately 74 persons who could be in the FB-Line facility during processing activities, about 56 would be in areas where they could receive substantial doses from a criticality. Of the 56, an estimated 4 workers could receive lethal doses of radiation, while the other individuals would receive varying nonlethal levels.

E.3.2.3 FA-Line

For accidents postulated for FA-Line, with the exception of a red-oil explosion or a severe earthquake, no substantial injuries to involved workers are likely. The force of the explosion or flying debris initiated by the red-oil explosion could result in physical injuries to involved workers. Although the likelihood for an involved worker fatality due to radiation exposure alone after a severe earthquake is minimal, the earthquake itself could result in significant injuries or death for involved workers.

E.3.2.4 235-F Storage Vaults

With the exception of an inadvertent nuclear criticality in the storage vaults, no fatalities to involved workers from the accident scenarios postulated for the 235-F facility are likely as a result of exposure to radiation. Section E.7 discusses the criticality safety program. Because the number of persons permitted in the 235-F storage vaults is limited, the number of individuals who could be impacted from an inadvertent nuclear criticality would be limited. No more than two involved workers would be likely to receive lethal doses of radiation, with a limited number of additional individuals receiving exposures significantly above the annual administrative limit established for routine operations. For other postulated accident scenarios for the 235-F facility, exposures to involved workers are likely to be within limits established for routine operations, even if the inventories of materials within the vaults increased as a result of stabilization of materials at other SRS facilities.

E.3.2.5 HB-Line Facility

Fatalities to involved or close-in workers from the accident scenarios postulated for full operation of the HB-Line facility are not a likely result of exposure to radiation. For many of the accidents, releases would be contained in the gloveboxes and filtered through the process system and canyon ventilation systems. Because the ventilation system flows from areas of lowest to highest radioactivity, and because releases flow through an exhaust stack after passing through a filtration system, the worker doses from these accidents are not likely to be substantially larger than those received during routine operations. For postulated accidents in which the release is not likely to remain in the ventilation system, such as a ground-level airborne release initiated by a severe earthquake, involved worker exposures would be unlikely to result in adverse health effects. An inadvertent nuclear criticality is not considered credible in the HB-Line, either during current or full operations, due to the forms and isotopes of the materials. Therefore, exposures or fatalities are not likely from inadvertent nuclear criticalities.

E.3.2.6 Uranium Solidification Facility

With the exception of an inadvertent nuclear criticality during processing, no fatalities to involved workers from the accident scenarios postulated for the Uranium Solidification Facility are likely as a result of exposure of radiation. Section E.7 discusses the criticality safety program. If an inadvertent nuclear criticality occurred, either during processing (criticality in a liquid) or packaging and storage (criticality in a powder), the radiation field generated by the criticality could lead to involved worker fatalities. However, DOE expects that the number of fatalities would be limited to two; additional individuals in the facility could receive doses that significantly exceeded their annual administrative exposure limits.

E.3.2.7 H-Area Receiving Basin for Offsite Fuel

No fatalities are likely to involved workers from the radiological accident scenarios postulated for the Receiving Basin for Offsite Fuel. This assessment assumes that the minimum worker distance from a criticality event would be the 3 meters (10 feet) of water covering the fuel in the Receiving Basin for Offsite Fuel. The attenuation of dose would be considerable due to the water shielding. The estimated prompt dose to a facility standing very near the basin would be 12 rem. A total dose of

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TC 30 rem over an 8-hour period could be attained due to inhalation of fission gases and direct radiation from the basin if a worker did not leave.

E.3.2.8 Reactor Disassembly Basins

No fatalities are likely to involved workers from the radiological accident scenarios postulated for the reactor disassembly basins. Worker doses for all postulated basin accidents would be minimal. This conclusion is based on the fact that the fuels and targets stored in each basin are maintained at a distance below the surface level of the water sufficient to minimize involved worker exposures. In addition, in events that involved a substantial loss of basin water after which fuels and targets could be exposed to the air (e.g., draindown of half the basin water or discharge of all basin water following a severe earthquake), sufficient time would be available to allow involved workers to take the precautions necessary to evacuate the area or implement other actions to minimize exposures.

E.3.2.9 Other Facilities

In addition to the facilities discussed above, M-Area buildings, the Savannah River Technology Center, and the TNX facility contain nuclear materials addressed by this EIS. The high-level waste tanks, Defense Waste Processing Facility, and a dry storage facility could also be involved in the handling of these nuclear materials. TC

No fatalities to involved workers from the accident scenarios postulated for M-Area are likely as a result of exposure to radiation. DOE anticipates that involved worker doses received from accidents would be minimal because the area serves as a storage vault for stable materials and involves only routine monitoring and maintenance activities.

No fatalities to involved workers from the accident scenarios postulated for the Savannah River Technology Center are likely as a result of exposure to radiation from accidents involving these materials, and DOE anticipates that involved worker doses received from accidents would be minimal. This conclusion is based on the very small amount of irradiated, aluminum-clad fuel assembly pieces, which would be a candidate for further stabilization in other facilities. The only alternative proposed for this material in the Savannah River Technology Center is No Action.

DOE anticipates no radiation-induced fatalities would result from accidents in the TNX facility or the waste tanks. The tanks in both areas store liquid radioactive materials and involve routine monitoring or remote transfers. The high-level waste tanks are in F- and H-Areas.

The Programmatic Spent Fuel Management Draft EIS (DOE 1995f) includes the dry storage of spent fuel at the SRS as an alternative. Based on assumptions of radioactive material dispersal within the storage vault and an operator exit time of 30 seconds, no fatalities from radiological consequences would be likely. As with most storage facilities, workers are not routinely present other than to support intermittent maintenance and monitoring.

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DOE examined close-in worker hazards resulting from the operation of the Defense Waste Processing Facility and its support facilities in the Supplemental EIS on the DWPF (DOE 1994g). With the exception of doses from an earthquake, the radiation doses to workers would probably be small and certainly below lethal doses. As listed in Tables E-4 through E-12, the earthquake initiator results in estimates for lethal doses to an uninvolved worker, generally considered to be farther from the radiation source than a close-in worker. Therefore, a severe earthquake could result in fatalities due to radiation in addition to deaths directly attributable to the earthquake damage.

E.3.3 STABLE MATERIALS

Although this EIS considers no alternatives other than Continued Storage (No Action) for stable materials, this section summarizes the accident analyses presented in the safety analysis reports for the facilities housing these materials. These documents discuss accident impacts for an uninvolved worker and the maximally exposed individual off the Site.

E.3.3.1 Postulated Radiological Accidents for the M-Area Reactor Materials Facilities

The primary purpose of the M-Area facilities was to manufacture fuel and target assemblies. The enriched uranium storage vault is constructed of reinforced concrete with walls and roof 30 centimeters (12 inches) thick. The four walls extend 1.8 meters (6 feet) into the ground and rest on 0.6-meter (2-foot)-thick footings. The storage vault was constructed to be a "maximum resistance" area [able to withstand a Fujita Intensity Five (F-5) tornado or a Modified Mercalli VIII (MM VIII) earthquake with little or no damage]. The SRS document explaining the limited continued operations in M-Area contains accident analyses for the facilities containing the nuclear materials addressed by this EIS. The bounding event for impact on the maximally exposed individual is an explosion in Building 320-M, which would result in a risk of 0.00014 rem per year and a latent cancer fatality projection of 0.00000007. For the uninvolved worker for the same event, the estimated risk would be 0.00044 rem per year and the latent cancer fatality projection would be 0.00000018. This accident is representative of bounding events related to the storage of a variety of materials for which further stabilization is not required. This group contains all material in the

Reactor Material Area, including miscellaneous depleted uranium and uranium metal, oxide, slugs, cores, sludges, enriched uranium residues, lithium aluminum control rods, spargers, targets, unirradiated Mark-22s with lithium target tubes, natural and enriched lithium metal in cans, Mark-16 and Mark-22 tubes, Mark-31 slugs, and neptunium targets. Stable material is stored in Buildings 313-M, 315-M, 320-M, 321-M, 322-M, and 341-1M. Unirradiated Mark-31 slugs (depleted uranium in aluminum housings) constitute most of the inventory. The No-Action Alternative is proposed for the materials currently stored in M-Area.

E.3.3.2 Postulated Radiological Accidents for Savannah River Technology Center

Nuclear material used or stored in the Savannah River Technology Center includes a small amount of americium and curium solution and targets; americium-241 scrap; depleted uranium slurry, metal, and oxide; enriched uranium sweepings; etc.

Under the No-Action Alternative, current research activities at the Savannah River Technology Center would continue, and DOE would continue to store equivalent amounts and types of material in Building 773-A laboratories and cells. These materials are generally stored in limited-quantity cans, bottles, or sample carriers. Most are contained further in laboratory hoods, gloveboxes, or cells. These items, or equivalent new sample quantities, would be in a safe stable form for storage for several years.

The Savannah River Technology Center Safety Analysis Report summarizes consequences from postulated accidents at the center involving areas that contain the materials listed above. The actual contribution to the accident scenarios from these materials would be negligible, but these events are bounding for all alternatives for stable materials (i.e., the No-Action Alternative). An earthquake with a magnitude of 0.2 times gravity poses the highest risk for the maximally exposed individual. The risk associated with this event would be 0.00023 rem per year and the latent cancer fatality projection would be 0.00000012. In the highly unlikely event that this accident occurred, it would cause a projected increase of 0.48 in latent cancer fatalities. From the same event, the uninvolved worker risk would be 0.0043 rem per year and the latent cancer fatality projection would be 0.0000017.

E.3.3.3 Postulated Radiological Accidents for the TNX Research Facility

The TNX facility is a "radiological facility," as determined by the quantity of nuclear material present TE (DOE 1992b). This hazard classification is the lowest for a facility that contains radioactive materials and requires no safety analysis report. This assessment does not summarize accident analyses for this facility because the impacts are bounded by those for several other facilities; only the No-Action Alternative would apply.

E.4 Postulated Accidents Involving Extremely Hazardous Substances

Because of the many types of materials and chemicals at the Site and the varying quantities of these materials in different locations, the analysis of potential accident scenarios involving hazardous materials was limited to substances categorized by the U.S. Environmental Protection Agency as "Extremely Hazardous Substances." as designated under the Emergency Planning and Community Right-to-Know Act of 1986 (40 CFR Part 355). Although materials not categorized as Extremely TE Hazardous Substances can affect the health and safety of workers and the public if released in sufficient quantities and forms, the Site has implemented programs in accordance with DOE Order requirements (e.g., DOE 1985, 1993, 1994k) that incorporate programmatic and management TE requirements of other government agencies, such as the Occupational Safety and Health Administration. While these materials might present hazards to workers or the public if accidentally released to the environment, their impacts are likely to be bounded by potential impacts from accidents involving Extremely Hazardous Substances; therefore, this appendix does not analyze them, with one exception, benzene. Due to the large quantities of benzene associated with alternatives TC involving the Defense Waste Processing Facility vitrification process, this appendix includes the hazard analysis from the DWPF Supplemental EIS (DOE 1994g) for completeness.

This section presents potential impacts from postulated chemical accidents at facilities that are or could be involved with safely managing or stabilizing SRS nuclear materials. For each area, it presents potential impacts of the bounding hypothetical chemical accident scenarios (as calculated using the method described in Section E.2.4).

Substances present in bulk quantities can, in some cases, be reduced or eliminated after stabilization of the associated nuclear material. In other cases (e.g., the Receiving Basin for Offsite Fuel), the TE chemicals support long-term facility functions independent of the interim management of the nuclear materials covered in this EIS. The accident consequences presented in this section assume a maximum chemistry inventory and are bounding for all alternatives.

E.4.1 POSTULATED CHEMICAL ACCIDENTS FOR F-AREA FACILITIES

Based on a review of current inventories at the facilities in the F-Area (DOE 1995f), DOE determined TE that seven Extremely Hazardous Substances are in use in the area. Table E-13 lists the maximum

Table E-13. Inventories of Extremely Hazardous Substances[a] in F-Area.

Substance	Maximum amount in a single location (kilograms)[b,c]
Hydrochloric acid	34.0
Hydrogen fluoride	1,174.8
Hydrogen peroxide	122.5
Nitric acid	65,771.6
Phenol	0.9
Phosphorous pentoxide	0.9
Sulfuric acid	3,823.8

a. Materials categorized as Extremely Hazardous Substances (40 CFR Part 355), as designated under the Emergency Planning and Community Right-to-Know Act of 1986.
b. To determine the quantity in pounds, multiply by 2.2046.
c. Amounts are based on 1993 (1-year) values.

TC amounts of each substance in a single location in the F-Area. These inventories are for the F-Area separations facilities and do not include the F- and H-Area Tank Farms.

To determine airborne concentrations at 640 meters (2,100 feet) and the nearest SRS boundary (the locations of the uninvolved worker and maximally exposed offsite individual, respectively), DOE assumed an inadvertent release to the environment of the maximum amount of each material in a single location. This method enables a comparison of the impacts of the various substances as well as impacts at the facilities housing these substances. These impacts are conservative because the analysis does not consider the frequency of an initiating event that could lead to the release of this maximum amount.

DOE used the EPICode™ computer code (see Section E.2.6) to model the release of each material. Table E-14 lists the results of the analyses and compares expected airborne concentrations at the uninvolved worker and maximally exposed individual locations to the different threshold Emergency Response and Planning Guidelines or their equivalents.

Because a severe seismic event has the potential to initiate the release of the same material from different locations in the F-Area, DOE analyzed a release of the maximum daily inventory. Table E-15 lists the results of these analyses. A total release of the entire inventory of a particular material from the F-Area to the environment is extremely unlikely, especially if the material is in several different locations, facilities, or buildings in the area. However, the assumption of a total release of the maximum inventories in the area provides a bounding estimate for the largest airborne concentrations DOE could expect following a severe earthquake.

Table E-14. Impacts from potential non-seismic-initiated releases of extremely hazardous substances in F-Area.

Substance released	Maximum amount in F-Area (kg)[a]	Airborne concentration (milligram per cubic meter)[b]				
		At 640m[c]	At Site boundary[d]	ERPG-1[e]	ERPG-2[e]	ERPG-3[e]
Hydrochloric acid	3.4E+01	6.3E-03	8.5E-05	4.5	3.0E+01	1.5E+02
Hydrogen fluoride	1.2E+03	2.2E+02	2.9	4.0	1.6E+01	4.1E+01
Hydrogen peroxide	1.2E+02	2.3E-02	3.1E-04	1.4	–	1.1E+02
Nitric acid	6.6E+04	1.4E+01	3.6	5.2	3.9E+01	7.7E+01
Phenol	9.1E-01	1.5E-04	1.7E-06	3.9E+01	1.9E+02	7.7E+02
Phosphorous pentoxide	9.1E-01	1.5E-04	1.7E-06	5.0	2.5E+01	1.0E+02
Sulfuric acid	3.8E+03	2.2E-07	3.7E-09	2.0	1.0E+01	3.0E+01

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a. To determine the quantity in pounds, multiply by 2.2046.
b. Airborne concentrations derived assuming conservative (99.5 percentile) meteorological conditions for the Site.
c. Location of the uninvolved worker, assumed to be located 640 meters (2,100 feet) downwind from the release.
d. Location of the maximally exposed offsite individual, assumed to reside at the nearest SRS boundary downwind from the point of release at 10.6 kilometers (6.6 miles).
e. Either the Emergency Response Planning Guidelines value or most restrictive exposure guideline available, as discussed in Section E.2.4 and listed in Table E-2. For substances with limits established in terms of parts per million, the value in milligrams per cubic meter was determined using the following equation: milligrams per cubic meter = (limit in parts per million) x (gram molecular weight of substance) / 24.45.

Table E-15. Impacts from potential releases of extremely hazardous substances in F-Area resulting from a severe earthquake.

Substance released	Maximum daily amount in entire F-Area (kg)[a]	Airborne concentration (milligram per cubic meter)[b]				
		At 640m[c]	At Site boundary[d]	ERPG-1[e]	ERPG-2[e]	ERPG-3[e]
Hydrochloric acid	1.0E+02	1.9E-02	2.6E-04	4.5	3.0E+01	1.5E+02
Hydrogen fluoride	1.2E+03	2.2E+02	2.9	4.0	1.6E+01	4.1E+01
Hydrogen peroxide	1.2E+02	2.3E-02	3.1E-04	1.4	–	1.1E+02
Nitric acid	2.7E+05	3.9E+02	1.4E+01	5.2	3.9E+01	7.7E+01
Phenol	1.4	2.3E-04	2.6E-06	3.9E+01	1.9E+02	7.7E+02
Phosphorous pentoxide	9.1E-01	1.5E-04	1.7E-06	5.0	2.5E+01	1.0E+02
Sulfuric acid	4.0E+03	2.3E-07	4.0E-09	2.0	1.0E+01	3.0E+01

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a. To determine the quantity in pounds, multiply by 2.2046.
b. Airborne concentrations derived assuming conservative (99.5 percentile) meteorological conditions for the Site.
c. Location of the uninvolved worker, assumed to be located 640 meters (2,100 feet) downwind from the release.
d. Location of the maximally exposed offsite individual, assumed to reside at the nearest SRS boundary downwind from the point of release at 10.6 kilometers (6.6 miles).
e. Either the Emergency Response Planning Guidelines value or most restrictive exposure guideline available, as discussed in Section E.2.4 and listed in Table E-2. For substances with limits established in terms of parts per million, the value in milligrams per cubic meter was determined using the following equation: milligrams per cubic meter = (limit in parts per million) x (gram molecular weight of substance) / 24.45.

As listed in Tables E-14 and E-15, the airborne concentrations for a gaseous release of hydrogen fluoride (hydrofluoric acid) would exceed the ERPG-3 threshold limits at 640 meters (2,100 feet) from the point of release. As explained in Section E.2.4, ERPG-3 threshold values represent concentrations at which an individual would experience or develop life-threatening health effects if exposed for longer than 1 hour. Because individuals could be notified and evacuated to a safe location (e.g., inside a building with adequate ventilation) within 1 hour of an inadvertent release of hydrogen fluoride, DOE does not expect any life-threatening or long-term health effects to uninvolved workers. Uninvolved workers could experience mild burning of the lungs from inhaling hydrogen fluoride, burning of the eyes, and mild skin irritations. In addition, because the airborne concentrations at the nearest SRS boundary would be below ERPG-1 threshold values, no measurable health effects are likely to members of the public. However, for involved workers, there is a potential for serious worker injury and potential fatalities because of the large concentrations expected at locations close to the point of release, which could hinder personnel from taking appropriate emergency response actions.

Table E-15 indicates that, in the event of a severe earthquake, a release of the total quantity of nitric acid in the F-Area would exceed ERPG-3 values at a distance of 640 meters (2,100 feet) and ERPG-1 values at the nearest SRS boundary. As explained in Section E.2.4, the health effects from being exposed to ERPG-1 threshold values for greater than 1 hour are minor (e.g., irritation of the eyes and objectionable odor). For uninvolved and involved workers, although the release would exceed ERPG-3 threshold values, no worker fatalities from exposure to airborne acid concentrations would be likely; some individuals could experience significant short-term health effects, such as burning of the lungs and irritation of the skin. Because this scenario assumes that all nitric acid in the F-Area would be released from a single location during a severe earthquake, airborne concentrations would be lower than those listed in Table E-15.

E.4.2 POSTULATED CHEMICAL ACCIDENTS FOR H-AREA FACILITIES

Based on a review of current inventories at the various H-Area facilities (WSRC 1994h), DOE determined that seven Extremely Hazardous Substances are in use in the H-Area. Table E-16 lists the maximum amounts of each substance in a single location in the H-Area. These inventories are for the H-Area separations facilities and do not include the F- and H-Area Tank Farms.

TC

Table E-17 lists the results of the analyses and compares the expected airborne concentrations at the uninvolved worker and maximally exposed individual locations to the different threshold Emergency

Table E-16. Inventories of Extremely Hazardous Substances[a] in H-Area.

Substance	Maximum amount in a single location (kilograms)[b,c]
Ammonia	27.2
Hydrochloric acid	2.7
Hydrogen fluoride	2.3
Nitric acid	39,814.7
Nitric oxide	1,315.4
Phosphorous pentoxide	1.4
Sulfuric acid	0.9

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a. Materials categorized as Extremely Hazardous Substances (40 CFR Part 355), as designated under the Emergency Planning and Community Right-to-Know Act of 1986.

b. To determine the quantity in pounds, multiply by 2.2046.

c. Amounts are based on 1993 (1-year) values.

Table E-17. Impacts from potential non-seismic-initiated releases of Extremely Hazardous Substances in H-Area.

	Maximum amount in a	Airborne concentration (milligram per cubic meter)[b]				
Substance released	single H-Area location (kg)[a]	At 640m[c]	At Site boundary[d]	ERPG-1[e]	ERPG-2[e]	ERPG-3[e]
Ammonia	2.7	5.1E-03	5.8E-05	2.5E+01	2.0E+02	1.0E+03
Hydrochloric acid	2.7	5.0E-04	5.7E-06	4.5	3.0E+01	1.5E+02
Hydrogen fluoride	2.3	4.3E-04	4.9E-06	4.0	1.6E+01	4.1E+01
Nitric acid	4.0E+04	9.5E+01	1.9	5.2	3.9E+01	7.7E+01
Nitric oxide	1.3E+03	4.9E+03	4.4	9.3E+01	1.2E+02[f]	1.2E+02
Phosphorous pentoxide	1.4	1.2E-01	1.1E-03	5.0	2.5E+01	1.0E+02
Sulfuric acid	9.0E-01	1.7E-04	1.9E-06	2.0	1.0E+01	3.0E+01

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a. To determine the quantity in pounds, multiply by 2.2046.

b. Airborne concentrations derived assuming conservative (99.5 percentile) meteorological conditions for the Site.

c. Location of the uninvolved worker, assumed to be 640 meters (2,100 feet) downwind from the release.

d. Location of the maximally exposed offsite individual, assumed to reside at the nearest SRS boundary downwind from the point of release at 10.6 kilometers (6.6 miles).

e. Either the Emergency Response Planning Guidelines value or most restrictive exposure guideline available, as discussed in Section E.2.4 and listed in Table E-2. For substances with limits established in terms of parts per million, the value in milligrams per cubic meter was determined using the following equation: milligrams per cubic meter = (limit in parts per million) × (gram molecular weight of substance) / 24.45.

f. Alternative concentration limit guideline for ERPG-2 value (TWA × 5) was adjusted down to the next higher range value (IDLH).

Response and Planning Guidelines or their equivalents. Because a severe seismic event has the potential to initiate the release of the same material from different locations within the H-Area, DOE analyzed a release of the maximum daily inventory to the environment. Table E-18 lists the results of these analyses.

Table E-18. Impacts from potential releases of extremely hazardous substances in H-Area resulting from a severe earthquake.

Substance released	Maximum daily amount in entire H-Area (kg)[a]	Airborne concentration (milligram per cubic meter)[b]				
		At 640m[c]	At Site boundary[d]	ERPG-1[e]	ERPG-2[e]	ERPG-3[e]
Ammonia	2.7E+01	5.1E-03	5.8E-05	2.5E+01	2.0E+02	1.0E+03
Hydrochloric acid	1.1E+01	2.1E-03	2.4E-05	4.5	3.0E+01	1.5E+02
Hydrogen fluoride	3.6	6.7E-04	7.6E-06	4.0	1.6E+01	4.1E+01
TE Nitric acid	1.2E+05	2.3E+02	5.7	5.2	3.9E+01	7.7E+01
Nitric oxide	1.3E+03	4.9E+03	4.4	9.3E+01	1.2E+02[f]	1.2E+02
Phosphorous pentoxide	1.4	1.2E-01	1.1E-03	5.0	2.5E+01	1.0E+02
Sulfuric acid	2.7	5.0E-04	5.7E-06	2.0	1.0E+01	3.0E+01

---

a. To determine the quantity in pounds, multiply by 2.2046.

b. Airborne concentrations derived assuming conservative (99.5 percentile) meteorological conditions for the Site.

c. Location of the uninvolved worker, assumed to be located 640 meters (2,100 feet) downwind from the release.

d. Location of the maximally exposed offsite individual, assumed to reside at the nearest SRS boundary downwind from the point of release at 10.6 kilometers (6.6 miles).

e. Either the Emergency Response Planning Guidelines value or most restrictive exposure guideline available, as discussed in Section E.2.4 and listed in Table E-2. For substances with limits established in terms of parts per million, the value in milligrams per cubic meter was determined using the following equation: milligrams per cubic meter = (limit in parts per million) × (gram molecular weight of substance) / 24.45.

f. Alternative concentration limit guideline for ERPG-2 value (TWA × 5) was adjusted down to the next higher range value (IDLH).

As listed in Tables E-17 and E-18, the airborne concentrations for a gaseous release of nitric oxide would exceed the ERPG-3 threshold limits at a distance of 640 meters (2,100 feet) from the point of release. Table E-18 indicates that, in a severe earthquake, a release of the total quantities of nitric acid in the H-Area would exceed ERPG-3 values at a distance of 640 meters (2,100 feet) and ERPG-1 values at the nearest SRS boundary. For uninvolved and involved workers, although the release would exceed ERPG-3 threshold values, no worker fatalities from exposure to the airborne acid concentrations would be likely; some individuals could experience significant short-term health effects, such as burning of the lungs and irritation of the skin. Because this scenario assumes that all nitric acid in the H-Area would be released from a single location during a severe earthquake, airborne concentrations would be lower than those listed in Table E-18.

E.4.3 POSTULATED CHEMICAL ACCIDENTS FOR K-, L-, AND P-REACTOR BASINS

Based on a review of the chemical inventory that supports the water chemistry in the L-Reactor basin,
TE (WSRC 1994h) DOE determined that the only identified Extremely Hazardous Substance was a small
quantity of nitric acid. For 45.6 kilograms (100 pounds) of nitric acid modeled as a liquid spill (the
maximum daily amount in the basin), the airborne concentration at 640 meters (2,100 feet) would be

several orders of magnitude lower than the ERPG-1 concentration limit. DOE assumed that this was typical for all SRS reactor basins that store nuclear material.

In addition, because the airborne concentrations at the nearest SRS boundary would be considerably below ERPG-1 threshold values, no measurable health effects to members of the public would be likely. No impacts would hinder involved workers from taking appropriate emergency response actions.

E.4.4 POSTULATED CHEMICAL ACCIDENTS FOR M-AREA FACILITIES

Based on a review of current inventories at the various facilities in the M-Area (WSRC 1994h), DOE determined that five Extremely Hazardous Substances are in use in the area. Table E-19 lists the maximum amounts of each substance in a single location in the M-Area. However, M-Area contains nuclear materials that require no further stabilization. Therefore, this EIS proposes no alternatives for the safe management of nuclear materials in M-Area. As a result, no further chemical accident analysis is required. | TE

Table E-19. Inventories of Extremely Hazardous Substances[a] in M-Area.

Substance	Maximum amount in a single location (kilograms)[b,c]
Hydrochloric acid	34.0
Hydrofluoric acid	2.27
Nitric acid	34,807.5
Phenol	2.27
Sulfuric acid	15,241.0

a. Materials categorized as Extremely Hazardous Substances (40 CFR Part 355), as designated under the Emergency Planning and Community Right-to-Know Act of 1986.
b. To determine the quantity in pounds, multiply by 2.2046.
c. Amounts are based on 1993 (1-year) values.

E.4.5 SAVANNAH RIVER TECHNOLOGY CENTER

Based on a review of current inventories at the various facilities in the Savannah River Technology Center (WSRC 1994h), DOE determined that eight Extremely Hazardous Substances are in use in SRTC facilities. Table E-20 lists the total annual maximum and average daily quantities of these substances based on 1993 (1-year) inventories. In addition, Table E-20 lists the maximum amounts of each substance in a single location in the SRTC. However, the Center contains nuclear materials that require no further stabilization. Therefore, this EIS proposes no alternatives for the safe | TE

Table E-20. Inventories of Extremely Hazardous Substances[a] in Savannah River Technology Center.

Substance	Maximum amount in a single location (kilograms)[b,c]
Ammonia	0.5
Hydrochloric acid	2,215.4
Hydrogen fluoride	38.1
Nitric acid	3,864.2
Nitric oxide	0.9
Phenol	4.5
Phosphorous pentoxide	3.18
Sulfuric acid	13.6

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a. Materials categorized as Extremely Hazardous Substances (40 CFR Part 355), as designated under the Emergency Planning and Community Right-to-Know Act of 1986.
b. To determine the quantity in pounds, multiply by 2.2046.
c. Amounts are based on 1993 (1-year) values.

management of the nuclear materials in SRTC facilities. As a result, no further chemical accident analysis is required.

E.4.6 POSTULATED CHEMICAL ACCIDENTS FOR THE TNX AREA

Based on a review of chemical usage in the TNX area, DOE determined that no chemicals in the area were required to support the continued safe management of nuclear materials. As a result, no further chemical accident analysis was performed for the TNX area.

E.4.7 POSTULATED CHEMICAL ACCIDENTS FOR THE F- AND H-AREA TANK FARMS

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The information in this section is based on the Supplemental EIS for the Defense Waste Processing Facility. This information represents the chemical hazards associated with the existing and expected waste tank inventory. The quantities and types of chemicals that would support the storage and treatment of solutions containing plutonium, neptunium, or other nuclear materials not previously discharged to these tanks could change. If analyses showed that the current envelope is not bounding, safety analyses would be performed in accordance with DOE Orders 5480.21 and 5480.23 before such actions could occur. Table E-21 lists the inventory of chemicals and Table E-22 summarizes the results of comparisons to established thresholds.

Table E-21. Hazardous chemical inventory (designated as extremely hazardous substances) for the waste tank farms.[a,b]

Chemical	Building	Maximum daily amount[c] (kilograms[d])	Average daily amount[c] (kilograms)
Sulfuric acid	241-84H	10.9	4.1
	241-84H	3.2	(e)
	280-1F	3,828.8	10.4
	280-1H	3,794.3	1,683.8
Ammonia	241-58H	0.9	0.9
	242-24H	13.6	6.8
Nitric acid	241-61H	42,620.9	22,679.9
	241-84H	3.6	(d)
(60 to 71%)[f]	241-84H	0.5	(d)
	241-84H	0.5	(d)
Hydrochloric acid	241-84H	8.2	4.5
(36 to 37%)[f]	241-84H	9.1	4.5
(2.0 molar solution)	241-84H	22.7	10.9
Phosphorous pentoxide	241-84H	0.45	0.45

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a. Source: WSRC (1994h).

b. Inventories for a specified chemical might be in more than one facility or in several places in the same facility.

c. Maximum and average daily amounts are based on 1993 data.

d. To convert kilograms to pounds, multiply by 2.2046.

e. Average daily amounts not available.

f. Percentage of the chemical in the indicated solution.

TC

Table E-22. Summary of hazardous chemical assessment accident analysis results for the waste tank farms (milligrams per cubic meter).[a]

Chemical released	Maximum daily amount (kilograms[b])	Airborne concentrations At 100 meters[c]	Airborne concentrations At SRS boundary	ERPG 1[d]	ERPG 2	ERPG 3
Nitric acid (Bldg. 241-61H)	42,620.9	8.3E+02	2.0E+00	5.2E+00	3.9E+01	7.7E+01
Phosphorous pentoxide (Bldg. 241-84H)	0.45	7.5E-02	3.1E-04	5.0E+00	2.5E+01	1.0E+02
Ammonia (Bldg. 242-24H)	13.6	4.5E-03	2.4E-05	1.7E+01	1.4E+00	7.0E+02
Hydrochloric acid [2.0 meter solution] (Bldg. 280-1H)	22.7	7.6E-03	3.9E-05	4.5E+00	3.0E+01	1.5E+02
Sulfuric acid (Bldg. 280-1F)	3,828.8	3.7E-06	3.2E-09	2.0E+00	1.0E+01	3.0E+01

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a. Source: WSRC (1994h).

b. To convert kilograms to pounds, multiply by 2.2046.

c. To convert meters to feet, multiply by 3.281.

d. Emergency Response Planning Guidelines.

E.4.8 POSTULATED CHEMICAL ACCIDENTS FOR THE DEFENSE WASTE PROCESSING FACILITY AND IN-TANK PRECIPITATION

The information in this section is based on the Supplemental EIS for the Defense Waste Processing Facility and represents the chemical hazards from the operation of the DWPF and In-Tank Precipitation facilities. The chemicals required or generated by the vitrification of different feed materials (e.g., a plutonium-bearing solution could be different). Technical studies to support the alternatives involving these facilities would be performed to determine if the hazards described in this EIS are bounding. Table E-23 and E-24 provide summary data for the vitrification facility and in-tank precipitation facility, respectively.

Table E-23. Summary of the vitrification facility chemical hazard comparisons (milligrams per cubic meter).[a]

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				Airborne concentrations				
Accident	Location	Initiator	Frequency (annual)	At 100 meters[b,c]	At SRS boundary	ERPG-1[d] (mg/m^3)	ERPG-2 (mg/m^3)	ERPG-3 (mg/m^3)
Benzene[e] Release	Organic Waste Storage Tank	Explosion	2.7E-04	1.4E+04	5.7E+00	1.6E+01	1.6E+02	9.6E+03
	Organic Waste Storage Tank	Tornado (176kph)[f]	1.0E-04	1.0E+04	1.5E+01	1.6E+01	1.6E+02	9.6E+03
Formic Acid Release (90 percent solution)	Cold Feed Area	Earthquake (0.1*g*)	2.0E-03	1.0E+02	6.0E-02	1.9E+01	2.9E+01	5.7E+01
	Chemical and Industrial Waste Treatment Area	Earthquake (0.1*g*)	2.0E-03	4.9E+01	3.0E-02	1.9E+01	2.9E+01	5.7E+01
Nitric Acid Release (50 percent solution)	Cold Feed Area	Tornado (176kph)[f]	1.0E-04	6.3E+01	3.0E-02	5.2E+00	3.9E+01	7.7E+01
	Cold Feed Area	Earthquake (0.1*g*)	2.0E-03	6.3E+01	5.0E-02	5.2E+00	3.9E+01	7.7E+01
	Vitrification Building	Leaks, transfer errors, overflows, etc.	4.8E-02	2.1E-03	2.4E-04	5.2E+00	3.9E+01	7.7E+01
	Chemical and Industrial Waste Treatment Area	Earthquake (0.1*g*)	2.0E-03	6.2E+01	3.0E-02	5.2E+00	3.9E+01	7.7E+01
Mercury Release (Vapor)	Melter Off-gas	Loss of cooling	(g)	3.7E-03	4.2E-04	1.5E-01	2.0E-01	2.8E+01

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a. Source: WSRC (1994h).
b. To convert meters to feet, multiply by 3.281.
c. Peak 15-minute-average airborne concentrations.
d. Emergency Response Planning Guidelines.
e. Suspected human carcinogen. Available epidemiologic studies are conflicting or insufficient to confirm an increased risk of cancer in exposed humans.
f. kph = Kilometers per hour; maximum wind speed; to convert kilometers to miles, multiply by 0.62137.
g. Because consequences would be negligible, frequency is not calculated for mercury releases.

Table E-24. Summary of in-tank precipitation accident analysis results (milligrams per cubic meter).[a]

		Airborne concentrations				
	Chemical	At 100 meters[b]	At SRS boundary	ERPG-1[c]	ERPG-2	ERPG-3
Sodium titanate (ST) tank spill	Sodium titanate	9.4E+00	1.5E-02	(d)	(d)	(d)
	Methanol	1.3E+01	2.1E-02	2.6E+02	1.3E+03	6.5E+03
	Isopropanol	2.0E+01	3.3E-02	9.8E+02	9.8E+02	2.9E+04
Sodium tetraphenylborate (STPB) tank spill	Sodium tetraphenyl borate	6.9E+01	1.1E-01	(e)	(e)	(e)
	Benzene	4.0E+02	6.4E-01	1.6E+01	1.6E+02	9.6E+03
Oxalic acid tank spill	Oxalic acid	2.6E+00	4.1E-03	2.0E+00	5.0E+00	5.0E+02
Caustic (sodium hydroxide) tank spill	Sodium hydroxide	1.1E-01	1.9E-04	2.0E+00	4.0E+01	1.0E+02
Benzene release due to chemical reaction	Benzene	5.8E+03	9.3E+00	1.6E+01	1.6E+02	9.6E+03
Nitrogen asphyxiation in stripper building	Air concentrations are not applicable. Nitrogen is used as the stripping gas to remove benzene from filtrate and wash waters. If the nitrogen leaked into the building in sufficient quantities, a worker could be subject to asphyxiation due to oxygen in the air.					

TC

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a. Source: WSRC (1994h).
b. To convert to feet, multiply by 3.281.
c. Emergency Response Planning Guidelines.
d. Guideline values for sodium titanate are unavailable.
e. Guideline values for sodium tetraphenylborate are unavailable.

E.5 Environmental Justice

When the 99.5 percent meteorology model is used, the SRS sector most affected by accidents is the Northwest. Although this is not typical of weather conditions (e.g., not the prevailing wind direction), the model calculated the highest impact to an individual at the SRS boundary.

Figures 3-7 and 3-8 show the distributions, by census tracts, of people of color and low-income populations, respectively. Parts of two census tracts in the Northwest sector adjoin the SRS. Neither tract is a low-income community or a community comprised of 50 percent or more of people of color, although one of the tracts contains between 35 and 50 percent people of color.

Farther from the SRS in the Northwest sector are low-income communities and communities that contain 50 percent or more of people of color. However, other communities in the sector are not low-income and contain fewer than 35 percent people of color, and they are as close as, or closer to, the SRS boundaries than the low-income communities or the communities of people of color.

Based on the distribution of types of communities and on the low dose received by the maximally exposed individual (see tables in this appendix), the accident scenarios would not result in disproportionately high or adverse human health and environmental impacts on people of color or low-income populations.

E.6 Accident Mitigation

Although DOE expends extensive efforts and large amounts of money to prevent accidents involving radioactive and hazardous materials, accidents and inadvertent releases to the environment can still occur. Therefore, an important part of the accident analysis process is the identification of actions that can mitigate consequences from accidents if they occur.[2] This section summarizes the SRS Emergency Plan, which governs responses to accident situations that could affect Site employees or the offsite population.

TE | The *Savannah River Site Emergency Plan* (WSRC 1994f) defines appropriate response measures for the management of SRS emergencies (e.g., radiological or hazardous material accidents). It incorporates into one document the entire process designed to respond to and mitigate the consequences of a potential accident. For example, it establishes protective action guidelines for accidents involving chemical releases to keep onsite and offsite exposures as low as possible. It accomplishes minimization or prevention of exposures by minimizing time spent in the vicinity of the hazard or the release plume, keeping personnel as far from the hazard or plume as possible (e.g., using physical barricades and evacuation), and taking advantage of available shelter.

Emergencies that could cause activation of all or portions of this plan and the SRS Emergency Response Office include the following:

- Events (operational, transportation, etc.) with the potential to cause releases above allowable limits of radiological or hazardous materials.
- Events (fires, explosions, tornadoes, hurricanes, earthquakes, dam failures, etc.) that affect or could affect safety systems designed to protect Site and offsite populations and the

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[2] This analysis takes no credit for accident response under the SRS Emergency Plan in determining the potential consequences and risks to workers or members of the public presented in other sections of this appendix.

environment. The effectiveness of the emergency plan would depend on the severity of the event and the impact on the Site and local infrastructure.

- Events (bomb threats, hostage situations, etc.) that reduce the security posture of the Site.
- Events created by proximity to other facilities such as the Vogtle Electric Generating Plant (a commercial nuclear utility across the Savannah River from the Site) or nearby commercial chemical facilities.

Depending on the types of postulated accidents and the potential impacts that could result from those accidents, emergencies are classified in several categories in accordance with requirements defined in the DOE 5500 Series of Orders, as follows:

- Alerts are confined within the affected facility boundary; no measurable impacts to workers or members of the public outside the facility boundary are likely.
- Site Area Emergencies are events that are in progress or that have occurred involving actual or likely major failures of facility safety or safeguards systems needed for the protection of onsite personnel, the public, the environment, or national security; because they have the potential to impact workers at colocated facilities or members of the public in the SRS vicinity, these situations require notification of and coordination of responses with the appropriate local authorities.
- General Emergencies produce consequences that require the implementation of protective actions to minimize impacts to both workers and the public; full mobilization of all available onsite and offsite resources is usually required to deal with the event and its consequences.

In accordance with the Site Emergency Plan, DOE conducts periodic drills and exercises at the SRS to develop, maintain, and test response capabilities, and validate the adequacy of emergency facilities, equipment, communications, procedures, and training. For example, drills occur for the following accident scenarios in the facilities or facility areas: facility or area evacuations, shelter protection, toxic gas releases, nuclear incident monitor alarms (following an inadvertent nuclear criticality), fire alarms, medical emergencies, and personnel accountability (to ensure that all personnel have safely evacuated a facility or area following an emergency). DOE and Westinghouse Savannah River Company conduct and evaluate periodic drills with the following organizations or groups to ensure that they continue to maintain (from both a personnel and an equipment standpoint) the capability to respond adequately to emergency situations: first aid teams; rescue teams; fire wardens, fire response and firefighting teams; SRS medical and Health Protection personnel and personnel from the Eisenhower Army Medical Center; SRS and local communications personnel and systems; SRS security forces; and SRS Health Protection agencies.

E.7 Nuclear Criticality Safety Program

TC

As discussed above, with the exceptions of an inadvertent nuclear criticality and a severe earthquake, no fatalities to involved workers from accident scenarios postulated for the management, stabilization, or storage of nuclear material would be likely to result from exposure to radiation.. Because an earthquake cannot be prevented, various physical and structural features are designed to mitigate the impacts. It is, however, possible to prevent criticalities. A criticality occurs when a neutron fissions the nucleus of a fissionable material to produce energy, fission fragments, neutrons, and various radiations. While nuclear reactors are specifically designed to produce energy from fission by controlling this neutron chain reaction, nonreactor nuclear facilities at the SRS do not generally provide the same control, shielding, and containment characteristics. Thus, an inadvertent fission chain reaction (nuclear criticality) in an SRS nonreactor nuclear facility could produce harmful radiation-related effects on nearby personnel.

As a result, nuclear criticality safety has been defined as "the prevention or termination of inadvertent nuclear chain reactions in nonreactor environments." In practice, the first concept--prevention--is by far the primary goal. As a consequence, SRS maintains a nuclear criticality safety program that establishes and defines the principles, practices, and controls to be used for the prevention of criticality accidents. When it has been determined that the potential for an inadvertent nuclear criticality accident exists for a facility, the design of criticality controls, including equipment and procedures, shall meet, at a minimum, the requirements described in the WSRC Nuclear Criticality Safety Manual. For a new facility, the use of physical design features to prevent criticality would be preferable. To ensure the successful implementation of this program, a training policy recently adopted at the SRS supports the goal that all reasonable efforts shall be taken to reduce or eliminate the potential for, and consequences of, a criticality accident. Nuclear criticality safety training programs at the SRS are developed to be consistent with DOE Orders 5480.20 and 5480.24 for operating facility personnel and all other personnel requiring criticality safety training.

Positive identification of fissionable material, particularly fissile material, is essential to criticality safety. Adequate labeling of fissionable material and clear posting of work and storage areas in which fissionable materials are present are important in avoiding the accumulation of unsafe quantities of such materials. Appropriate fissionable material labeling and area posting are maintained at SRS nonreactor nuclear facilities specifying material identification and all parameter limits subject to procedural control. Storage requirements include minimum spacing distances to prevent sufficient material from being in close proximity. Criticality "poisons," such as boron, are often used in storage racks or packaging for material.

Written plans and procedures govern operations at SRS in which criticality safety is a consideration. These plans and procedures cover startup, operations, and any modifications that might affect criticality safety. Procedures clearly specify all controlled parameters and limits related to criticality safety. All criticality safety-related limits contained in the operating procedures are based on Nuclear Criticality Safety Evaluations (NCSEs). New or revised procedures containing nuclear safety steps, criticality safety limits, or criticality safety requirements undergo review and approval by a Criticality Safety Engineering Group (CSEG) before implementation. In the event of a criticality limit violation, SRS procedures specifically govern actions to be taken in the event of an undesirable situation; the objective of such procedures is to place the operation into as stable and safe a condition as possible until a criticality safety engineer or specialist can conduct an evaluation.

Water, the most often used firefighting agent, is an efficient moderator and reflector of neutrons (i.e., it can contribute to a criticality). In the absence of moderating materials such as water, relatively large masses of dry fissile materials such as powders or metals can be handled safely. In the event of a fire, SRS nonreactor nuclear facilities maintain prefire plans prepared by the management and engineering staff of each facility with the assistance by the Criticality Safety Engineering Group, SRS fire safety engineers, and the Area Fire Department, as necessary. These plans help provide a framework for the successful combination of firefighting and criticality safety. The CSEG approves the prefire plans for each facility in which criticality safety is of concern.

The SRS maintains criticality alarm systems, or Nuclear Incident Monitors (NIMs). The primary purpose of NIM systems is to minimize, by means of quick detection and alarm, the acute dose received by personnel from a criticality (and potential recriticality) accident in areas where the cumulative absorbed dose in free air might exceed 12 rads. The secondary purpose of the NIM system is to notify people to stay clear of the evacuated area and to notify appropriate response teams.

Emergency procedures for criticality accidents are prepared for each SRS facility in which criticality safety controls are instituted or criticality alarm systems are installed. Such emergency plans are

approved by the appropriate management and the cognizant Criticality Safety Engineering Group,
TE and consistent with the Site Emergency Plan (WSRC 1994f).

E.8 Accident Descriptions

The larger facilities contain a variety of processes, equipment, and techniques used depending on the intended function. In determining the source terms for use in accident analysis, DOE examined the appropriate process or section of a facility for the specific material and adjusted the source term to correspond where necessary. The tables in Section E.3 list the "modes" or conditions to reflect the selection for that material. The following paragraphs explain the accident titles used in the tables in Section E.3.

<u>Unpropagated fire</u> – A fire that has localized impact and does not spread. It can be caused by ignition of flammable solvent, spontaneous burning of plutonium metal exposed to oxygen, or other causes. Radioactive particulates are dispersed in the immediate area of the fire and some might be released to the environment (e.g., during a filter fire). The fire lasts for a short period because the amount of combustible material is limited.

<u>Inadvertent transfer</u> – An unplanned transfer of a solution or liquid to an unintended location due to personnel error. The usual causes of such accidents are incorrectly installed piping connections or overflows from a vessel into a sump resulting from human errors.

<u>Coil and tube failure</u> – Some process vessels and tanks have internal coils for cooling or heating the stored solutions. The coils usually contain water or steam. The pressure inside the coils is normally higher than the pressure in the vessel. Should the coils leak or fail their internal pressure could be lost, resulting in radioactive solution entering the cooling water (or steam) system. If the leak is undetected, the contaminated water could be released through the system to the atmosphere without treatment.

<u>Inadvertent criticality</u> – These events are discussed in Section E.7.

<u>Severe earthquake</u> – An earthquake that would be expected every 5,000 years. The severity or magnitude is based on an assumed horizontal ground acceleration of 20 percent of the acceleration due to gravity. An earthquake of this magnitude could result in structural damage and a loss of confinement of nuclear materials.

Rupture storage container – Certain radioactive materials can cause a buildup of gases inside the container in which they are stored (e.g., metal can) if it contains organic materials (e.g., plastic bags). Other materials (e.g., plutonium metal) can oxidize and gain moisture if the container is not completely airtight. Eventually, the pressure buildup can cause the storage container to bulge or rupture. This could disperse the material in the area around the container and result in exposure of a worker performing routine surveillance.

Eructation – A thermal or chemical reaction causes material to spew from its container. This could be an energetic event resulting in localized contamination. For the materials discussed in this EIS, such events would occur inside the canyons and no workers would be directly affected.

Red-oil explosion – So named because the substance causing the explosion is a thick red liquid produced by the inadvertent addition of organics to a high nitrate solution. The event can be very energetic and can result in a sudden localized explosion. The radiological consequences would probably be confined to areas within the canyon facilities.

Tornado – A tornado exerts pressure due to high wind speed on the surfaces of a structure. The resulting damage could cause releases of stored materials within the structure or could disperse materials stored in pads.

Uncontrolled reaction – Adjustments are routinely made to solutions to produce a reaction under known controlled conditions. If an adjustment (e.g., adding acid) or a change in condition (e.g., heating the contents) produces unexpected or rapid reaction, that reaction is "uncontrolled." The energy from this type of reaction could cause radioactive solutions to overflow or erupt outside the tank in which they are stored.

Propagated fire – A fire that goes beyond the area of ignition. For the materials discussed in this EIS, a propagated fire does not self-extinguish. For example, it might spread from a glovebox into the surrounding room or other areas of the facility.

Basin overflow/draindown – An unplanned movement of water, either into the reactor basins (causing an overflow) or from the reactor basins (draindown), which results in a flow of the basin water to sumps or storm drains and into the Savannah River. Basin overflow would normally be caused by human error; basin draindown could be caused by a breach of the basin integrity due to an earthquake.

Hydrogen explosion – Hydrogen gas is generated by radiolysis when water is in a tank or can with nuclear materials. If a sufficient quantity of the atmosphere in the container is hydrogen, the gas can detonate or explode, rupturing the container and releasing nuclear material.

Energetic event – Energetic events cause penetration of the primary confinement barrier and, if sufficiently energetic, can result in the bypass of a secondary barrier. Medium energetic events include a cabinet fire, an uncontrolled reaction, and criticality.

TC Melter spill - Radioactive molten glass is released from the melter to the spill pan in the melt cell in the vitrification facility. This accident assumes that a fraction of the spilled molten glass becomes airborne, releasing radionuclides to the atmosphere.

APPENDIX F. PUBLIC COMMENTS AND RESPONSES

The U.S. Department of Energy published the Draft Interim Management of Nuclear Materials Environmental Impact Statement (DOE/EIS-0220D) in March 1995. The Environmental Protection Agency announced the availability of the Draft EIS for public review and comment in the *Federal Register* on March 17, 1995 (60 FR 52, pages 14432-14433); this announcement initiated the 45-day comment period, which ended May 1, 1995. This appendix includes the comments made by government agencies, private organizations, and individuals during the public comment period, and DOE's responses to those comments.

This appendix presents the comment letters (Comments L1 through L14) DOE received, the transcript of the single voice-mail comment received (Comment V1), and the transcripts of the oral presentations (Comments H1 through H7) made at the public hearings. This appendix also presents the DOE responses to these comments, as described below. All of the letters and comments are presented in the order in which they were received. DOE has responded to those that requested clarification and those that resulted in changes to the text of the EIS. Comment letter L6 includes evaluations submitted by agencies of the State of South Carolina responsible for reviewing environmental impact statements. Because these agencies had no comments on the EIS, DOE has presented that letter without including a response.

DOE considered all the comments it received during the public comment period in the preparation of this Final EIS. Where appropriate, DOE has revised the EIS in response to comments. In such cases, the revision is indicated in the margin of the page with a change bar and the comment number that prompted the revision.

DOE held an afternoon and an evening meeting to receive oral and written comments and to exchange information with the public on the Draft EIS in Savannah, Georgia, on April 11, 1995, and in North Augusta, South Carolina, on April 13, 1995. Eleven people attended these meetings in Savannah; 42 attended the North Augusta meetings. During the comment period, seven people presented formal comments at the four public meetings. This appendix summarizes some comments presented during discussions with DOE (see Comments H2 and H5). The complete transcripts of these meetings are available in the DOE Public Reading Rooms in the library of the University of South Carolina-Aiken Campus and the Forrestal Building in Washington, D.C. These discussions included questions and answers on the following topics related to the interim management of nuclear materials:

- Clarifying the scope of materials included
- The costs of the alternatives
- The cost and schedule of the EIS preparation
- Clarifying the basis for the 10-year period and analysis
- Clarifying the americium and curium vitrification alternatives and the technical basis for the preferred alternative
- The resolution of the unreviewed safety question concerning the integrity of the stack liner if a seismic event were to occur
- The technical issues associated with the alternatives in terms of waste generation and the potential for recycling materials
- The technical basis for categorization of the irradiated fuel stored in the Receiving Basin for Offsite Fuels (RBOF) as stable
- The technical basis for vitrification of americium and curium as the preferred alternative
- Clarifying the relationship of proposed actions and decisions from this environmental impact statement with those from other EISs on potential missions and activities
- The sensitivity of the information on nuclear materials, quantities, and storage locations contained in this environmental impact statement, as it would relate to potential theft, diversion or sabotage.
- Clarifying the significance of facility risks and health effects to the community surrounding the Savannah River Site
- Providing basic information on materials, hazards, and health effects

DOE has made this Final EIS available for review in DOE Public Reading Rooms in Washington, D.C., and Aiken, South Carolina, and has distributed it to individuals, organizations, and Federal, state, and local officials who commented on the Draft EIS and to others who have expressed interest in the National Environmental Policy Act process. DOE appreciates the time and effort of everyone who participated in this process.

Tables F-1, F-2, and F-3 list the government agencies, private organizations, and individuals, respectively, who submitted comments.

Table F-1. Government agencies commenting on draft environmental impact statement.

Comment	Agency	Representative
L1	United States Department of Commerce, National Oceanic and Atmospheric Administration, National Marine Fisheries Service, Southeast Regional Office	Andreas Mager, Jr.
L6	State of South Carolina, Office of the Governor[a]	Rodney P. Grizzle
L8	Chatham County-Savannah Metropolitan Planning Commission	David A. Rutherford, Harry Jue
L7	U.S. Environmental Protection Agency, Region IV	Heinz J. Mueller
L10	Department of the Air Force	Jeffery K. K. Ching
L14	Oak Ridge National Laboratory	E. D. Collins

a. The Governor's Office conducted an intergovernmental review of this EIS; offices participating included the Department of Labor, Department of Agriculture, Office of Community Grant Program, South Carolina Human Affairs Commission, Wildlife and Marine Resources Department, Department of Health and Environmental Control, and Department of Transportation.

Table F-2. Private organizations commenting on draft environmental impact statement.

Comment	Organization	Representative
L11	Natural Resources Defense Council and Energy Research Foundation	Andrew P. Caputo
L13	Wayne State University - Harper Hospital, Radiation Oncology Center	James Fontanesi, M.D., Jacek Wiezbicki, Ph.D.
H3	Citizens for Environmental Justice	Qasimah P. Boston

Table F-3. Individuals commenting on draft environmental impact statement.

Comment	Name
L2, H2	George Minot[a]
L3, H7	Charles R. Goergen[b]
L4	Richard W. Benjamin
L5	Robert H. Wilcox
L9	Philip H. Permar
L12	William A. Lochstet
V1	Wayne Sievers
H1	Fred Nadelman
H4	Karl D. Tesch
H5	Robert F. Overman
H6	Perry Halton

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a. The commentor addressed DOE with both written and verbal comments.

b. Public comment was made as a letter and a verbal statement at one of the public hearings. DOE has addressed the comments, which are the same in both forms, in the response to Comment L3.

UNITED STATES DEPARTMENT OF COMMERCE
National Oceanic and Atmospheric Administration
NATIONAL MARINE FISHERIES SERVICE
Southeast Regional Office
9721 Executive Center Drive N.
St. Petersburg, Florida 33702

April 4, 1995

Arthur B. Gould, Jr. Director
U.S. Department of Energy
Environmental Compliance Division
Savannah River Operations Office
P.O. Box 5031 Attn: DOE/EIS-0220D
Aiken, South Carolina 29804-5031

Dear Mr. Gould:

The National Marine Fisheries Service (NMFS) has reviewed the Draft Environmental Impact Statement (DEIS) for the Savannah River "Interim Management of Nuclear Materials." Based on our review, the document sufficiently addresses potential impacts to resources for which we have stewardship responsibilities. Although we are concerned over the possibility of accidental releases associated with handling and treatment of highly toxic materials, it appears that great effort has been devoted to containment. We note that the planned action is not expected to cause elimination or adverse impacts to wetlands or significant diminution in the quality of surrounding aquatic systems. It is also noted that significant increases in chemical discharges, including radiological materials, are not expected and that the discharges will not exceed State water quality parameters for acceptability.

Several agencies, including the NMFS, U.S. Fish and Wildlife Service, U.S. Army Corps of Engineers, U.S. Environmental Protection Agency, and the States of Georgia and South Carolina are jointly and individually examining aquatic resource protection and restoration needs in the Savannah River. These efforts have been initiated as a result of increasing concern over the River's environmental quality and growing recognition of its enormous fishery, natural aesthetic, recreational, power production, and other public interest features. Of particular interest to the NMFS and other agencies is the River's function as a spawning and nursery site for anadromous fishes that include American shad (Alosa sapidissima), blueback herring (Alosa aestivalis), striped bass (Morone saxatilis), Atlantic sturgeon (Acipenser oxyrhynchus) and shortnose sturgeon (Acipenser brevirostrum). Because of their migratory nature, these species utilize significant portions of the river including sections that would be impacted by discharges from the Savannah River Site. Accordingly, any modification in the selected alternative that could potentially affect these resources should be disclosed. L1-1

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Comment L1. Page 1 of 2.

Finally, in accordance with the Endangered Species Act of 1973, as amended, it is the responsibility of the appropriate Federal regulatory agency to review its activities and programs and to identify any activity or programs that may affect endangered or threatened species or their habitat. If it is determined that these activities may adversely affect any species listed as endangered or threatened, formal consultation with our Protected Species Management Branch must be initiated. The appropriate contact person for matters pertaining to protected species is Mr. Charles Oravetz who may be contacted at the letterhead address.

We appreciate the opportunity to provide these comments.

Sincerely,

[redacted]

Andreas Mager, Jr.
Assistant Regional Director
Habitat Conservation Division

Comment L1. Page 2 of 2.

Response to comment letter L1:

L1-1

DOE has not modified any of the alternatives described in the Draft EIS. DOE has included additional alternatives in the Final EIS, but none is likely to have impacts on threatened or endangered species because they would involve the use of existing SRS facilities or processes. If any modifications to the alternatives are required before implementation, DOE will review the changes to determine the environmental impacts. This is consistent with Departmental policy.

GEORGE M. MINOT
3 Bateau Road
Hilton Head Island, SC 29928-3012
803-363-5150

Memorandum

To: NEPA Compliance Officer, DOE-SR Operations Office
From: George M. Minot [redacted]
Date: April 11, 1995
Subject: **DEIS for Interim Management of Nuclear Materials at SRS**

L2-1 1. **Section 1.3** (Categories of Nuclear Materials) states "*the scope of the EIS does not include two types of nuclear material currently in the SRS inventory -- tritium and plutonium-238.*" However, the list of SRS nuclear materials in Table S-1 (see page S-4) does not include approximately 80,000 gallons of plutonium-239 solutions currently stored at the F-Canyon. Undoubtedly these plutonium-239 solutions are the subject of the separate EIS mentioned in **Section 1.6 (Related National Environmental Policy Act Documents).** Is this a correct assumption?

L2-2 2. **Table S-2 (Alternatives for the management of SRS nuclear materials)** indicates that the "*preferred alternative for managing Americium and Curium is vitrification in F-Canyon.*" According to Section 2.2 (Programmatic Material), "*DOE has evaluated an alternative for each material that would convert it to a stable and storable form for future use in DOE programs.*" Is vitrification the most practical temporary storage/transfer medium? Also, according to Chapter 7, adoption of this alternative will require "*modifications to produce the F-Canyon Vitrification Facility*", an irreversible or irretrievable commitment of an unknown amount of resources that will take at least 3 1/2 years to complete. Therefore, it would seem prudent for DOE to adopt the *Processing to Oxide* alternative (which I understand is the safest form).

Also, inasmuch as a decision has already been made to take expedited action to stabilize
the Plutonium-239 solutions currently stored at the F-Canyon site by processing them to
L2-3 metal, why not process the Americium and Curium to metal also? If that
alternative is not practical, why not consider transferring the 3,800 gallons of
Americium and Curium solutions via truck or rail car to the nearby Defense
L2-4 Processing Waste Facility (S-Area facility which has just been approved for
completion and operation) for vitrification rather than expending resources
to produce the F-Canyon Vitrification Facility?

Comment L2.

Response to comment letter L2:

L2-1

DOE has modified Chapter 1 in response to this comment. The commentor is correct in noting that tritium and plutonium-238 were not included in this EIS because these materials are covered in separate NEPA documents prepared by DOE. Similarly, the plutonium-239 solutions in F-Canyon have been discussed in a separate NEPA document and are not included in this EIS. Section 1.6 of this EIS describes these documents.

L2-2

DOE has modified Chapter 2 in response to this comment. DOE selected the Vitrification (F-Canyon) Alternative over the Processing to Oxide Alternative for the americium and curium solutions because of the difficulty associated with processing these particular materials to oxide. Americium and curium emit enough gamma radiation to make remote handling necessary for either stabilization method. However, the Vitrification (F-Canyon) Alternative would involve fewer manual steps, so radiation exposures to workers, in general, probably would be lower and there would be a lower probability of an accident. Moreover, the modifications needed to convert a portion of F-Canyon to provide a vitrification capability would be relatively smaller than the modifications needed to provide F-Canyon with the capability to produce oxide.

L2-3

DOE did not consider Processing to Metal as a viable alternative because the capability does not exist at the SRS or elsewhere to produce these elements as a metal on the scale necessary for this material.

L2-4

DOE evaluated transfer of the americium and curium solutions to the Defense Waste Processing Facility for vitrification through the high-level waste tanks. The DWPF was designed to receive only liquid solutions from the high-level waste tanks. Modifications to the DWPF to accept the direct transfer of solutions would be a major undertaking that DOE probably could not complete in the next 10 years. DOE has modified Chapter 2 to evaluate the transfer of the americium and curium solutions to the DWPF through the high-level waste tanks.

IMNM EIS Public Comment
Charles R. Goergen
510 Boardman Road
Aiken, SC 29803

I think the aluminum clad materials currently stored in the Receiving Basis Offsite Fuel (RBOF) should be considered for the same treatment options as the SRS reactor basin materials. The DNFSB recommended to the Secretary of Energy in Recommendation 94-1 "That preparations be expedited to process deteriorating irradiated reactor fuel stored in basins at the Savannah River Site into a form suitable for safe interim storage until an option for ultimate disposition is selected."

L3-1

Aluminum clad materials were not designed for indefinite storage. The fact that the RBOF material currently shows no signs of deterioration does not preclude future deterioration or its inherent susceptibility. I suggest that stabilization of RBOF stored, aluminum clad materials in the same manner as the reactor basin materials should be considered for the following reasons:

- It will preclude future vulnerability possibilities.
- It will free more RBOF storage space for future receipt of other materials such as Foreign Research Reactor Fuel.
- Stabilization by dissolution can eliminate the direct nuclear proliferation concerns if the Highly Enriched Uranium (HEU) material is isotopically diluted to Low Enriched Uranium (LEU).
- Recovered HEU when diluted to LEU has the possibility of being sold commercially, providing income for the US government.
- It can allow more economical use of taxpayer facilities by concurrent and consistent treatment for similar materials preventing a proliferation of material forms.
- Stabilization facility capabilities used for reactor materials may not exist after the 10 year "Stable Material" window.

L3-2

I also think that unirradiated reactor component materials such as depleted uranium or neptunium targets should be able to be treated in the same manner as the irradiated material. This could allow for "cold run" testing materials for stabilization processes and allow consolidation of similar types of material. The economic benefit of reducing the variety of existing forms should be a consideration.

Comment L3.

Response to comment letter L3:

L3-1

The Draft EIS categorized all aluminum-clad fuel stored in the Receiving Basin for Offsite Fuel (RBOF) as "stable" because there had been no observed degradation of this fuel. Since the publication of the Draft EIS, the Defense Nuclear Facilities Safety Board has stated that some of the material in RBOF has "significant corrosion," as discussed in Section 1.3. DOE placed fuel previously used in the Taiwan Research Reactor (TRR) and slugs from the Experimental Breeder Reactor (EBR-II) program in cans before placing that material in RBOF because of the poor condition of the fuel. DOE knew the material was degraded before placing it in RBOF, and repackaged it as an interim measure. DOE believed this repackaging would be sufficient for interim storage until it could make disposition decisions. However, it now appears that some of the canisters in which the TRR fuel and EBR-II slugs were placed might be releasing radioactivity to the basin water because gas evolution has been observed from some canisters. While gas evolution does not prove the release of radioactivity to the basin water, it does suggest that the integrity of the particular canisters might not be intact. DOE does not believe that any substantial corrosion can be attributed to the RBOF water chemistry, which is still of high quality. However, given the known poor condition of the material inside these canisters and the suspect condition of the canisters themselves, DOE has decided to categorize the TRR fuel and EBR-II slugs as candidates for stabilization. Chapter 2 now discusses applicable alternatives for the TRR fuel and EBR-II slugs, and Chapters 4 and 5 and Appendixes D and E now include data on this material group.

L3-2

The Draft EIS recognizes that unirradiated depleted uranium targets can be used in "cold run" operations for system qualification (see Section 2.3.4). The Draft EIS also includes unirradiated neptunium-237 targets for consolidation of material (see Section 2.2.3).

Comments on the Draft Environmental Impact Statement on the Interim Management of Nuclear Materials

Richard W. Benjamin
805 Hammond Drive
North Augusta, SC 29841
Phone: (803) 279-2560

This draft EIS is, in general, considerably better done than most I have reviewed recently. The problems identified are real, there is an appropriate preferred alternative, and the solutions proposed can be accommodated with existing facilities.

L4-1 **All of the materials are dealt with properly, except for the research reactor Spent Nuclear Fuel (SNF) stored in the Receiving Basin for Offsite Fuels (RBOF). This aluminum-based, highly-enriched spent fuel should not be included with the <u>STABLE</u> nuclear materials.**

The Research Reactor SNF stored in RBOF and the Irradiated Mark-16 and Mark-22 fuel stored in the reactor basins have the same constituents:

- highly enriched uranium (HEU)
- aluminum
- fission products.

These fuels are not stable in water unless the water is very, very clean. RBOF water is very, very clean and can be kept that way. Reactor basin water is not very clean and, because of the construction materials, cannot be made as clean as the water in RBOF. Therefore, the Research Reactor SNF stored in RBOF is in very good condition, while the Irradiated Mark-16 and Mark-22 fuel stored in the reactor basins is corroding.

DOE is currently planning to have a lot of Foreign Research Reactor SNF sent to South Carolina for storage at the Savannah River Site, much more SNF than can be stored in RBOF. Therefore, some of this Research Reactor SNF will have to be put in the reactor basins, where it will corrode. This is true even if dry storage facilities (which are no longer planned or funded) are built, for they cannot be built in time to avoid the problem.

The best solution to these problems is a strategy that offers a technically, politically, and economically sound approach for disposing permanently of aluminum-based, HEU SNF. It depends upon using both the US processing facilities at SRS and the smaller UK processing facilities at Dounreay to process and blend HEU-aluminum fuels to commercial-grade LEU under IAEA safeguards. This ends the proliferation and accidental criticality risks of HEU spent fuel and puts the recovered uranium and the high level waste into acceptable forms for commercial utilization and disposal, respectively.

I suggest the following integrated approach for HEU-Aluminum SNF disposition:

L4-2

I. The US Government establishes agreements and protocols with the foreign and domestic research reactor operators, the European Community, and the British Processing Facility at Dounreay which will:

(a) return most existing HEU fuels to the US for disposal by year 2003.

(b) make arrangements for the future (post 2003) supply and disposal of reduced enrichment fuels (non-weapons grade uranium), for both domestic and foreign reactors, through US and European nuclear fuel service organizations.

Comment L4. Page 1 of 2.

(c) arrange for the processing, under IAEA safeguards at Dounreay, of some of the foreign research reactor HEU spent nuclear fuel. Separated HEU would be blended to commercial-grade LEU in the facility so that it would no longer present a proliferation risk and could be used in current commercial reactors. L4-2

2. Use existing DOE processing and waste treatment facilities at SRS to treat all aluminum-based enriched uranium spent fuels stored at DOE sites and returned from research/test reactors. Assurance that HEU would not be used for weapons would be made an inherent part of the chemical processing operations by blending HEU solution with natural or depleted uranium to create a low enriched uranium (LEU) product. The relatively small high level waste (HLW) stream would be treated in existing waste processing facilities with the existing 30 million gallons of high level waste (HLW) at SRS and converted into stainless steel encapsulated borosilicate glass waste forms for geologic disposal. When the backlog of HEU spent fuel is completed, SRS processing facilities would be shut down and the remaining small annual quantity of research reactor HEU SNF processed at Dounreay (Dounreay is much smaller than the SRS facility and can treat the small annual throughput more economically).

Benefits

If this integrated approach were implemented, it would:

- dispose of waste products and weapons-capable uranium in SNF using existing technologies and eliminate prolonged storage. L4-3
- treat highly radioactive fission products with existing high level SRS waste and vitrify it into borosilicate glass waste forms in DWPF.
- isotopically dilute highly enriched uranium <u>during</u> processing, creating low enriched uranium (LEU) for use in power reactor fuels.
- comply with US non-proliferation goals now and in the future.

Advantages

This approach has several advantages over other alternatives for disposing of enriched uranium-aluminum spent fuels, including:

- significantly reduces uncertainties relative to other alternatives, for it is the quickest route to an already acceptable waste form for geologic disposal and commercial re-use of the uranium. There is still no assurance that HEU spent fuel will ever be acceptable for geologic disposal.
- provides the smallest number of HLW canisters for geologic disposal.
- carries the lowest capital investment.
- probably provides the lowest net (total) cost.
- provides no additional significant safety, health or environmental impacts.

Comment L4. Page 2 of 2.

Response to comment letter L4:

L4-1

The Draft EIS categorized all aluminum-clad fuel stored in the Receiving Basin for Offsite Fuel as "stable" because there had been no observed degradation of this fuel. Since the publication of the Draft EIS, the Defense Nuclear Facilities Safety Board has stated that some of the material in RBOF has "significant corrosion," as discussed in Section 1.3. DOE placed fuel previously used in the Taiwan Research Reactor and slugs from the Experimental Breeder Reactor program in cans before placing that material in RBOF because of the poor condition of the fuel. DOE knew the material was degraded before placing it in RBOF, and repackaged it as an interim measure. DOE believed this repackaging would be sufficient for interim storage until it could make disposition decisions. However, it now appears that some of the canisters in which the TRR fuel and EBR-II slugs were placed might be releasing radioactivity to the basin water because gas evolution has been observed from some canisters. While gas evolution does not prove the release of radioactivity to the basin water, it does suggest that the integrity of the particular canisters might not be intact. DOE does not believe that any substantial corrosion can be attributed to the RBOF water chemistry, which is still of high quality. However, given the known poor condition of the material inside these canisters and the suspect condition of the canisters themselves, DOE has decided to categorize the TRR fuel and EBR-II slugs as candidates for stabilization. Chapter 2 now discusses applicable alternatives for the TRR fuel and EBR-II slugs, and Chapters 4 and 5 and Appendixes D and E now include data on this material group.

L4-2, L4-3

Several ongoing DOE National Environmental Policy Act evaluations address the disposition of domestic and foreign highly enriched uranium aluminum-clad spent nuclear fuel. The *Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement* evaluates where the Department should manage these fuels among its candidate sites and proposes that it manage aluminum-clad fuel at the SRS. In its Record of Decision for the Programmatic EIS, issued on June 1, 1995, DOE decided to consolidate aluminum-clad fuel at the SRS. DOE did not announce a decision on the technology it would use to manage the spent fuel assigned to the SRS. As a result, and with regard to the disposition of domestic highly enriched uranium spent nuclear fuel, DOE probably will issue additional NEPA evaluations to determine a path for the management of this fuel.

The *Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel EIS* evaluates options in regard to the management of U.S.-origin aluminum-clad spent nuclear fuel that was used in foreign research reactors and is currently overseas. These options include returning the fuel to the United States, and continuing to manage the fuel overseas in part by returning the fuel to the United States and blending the highly enriched uranium to low enriched uranium.

In all of these evaluations, the United States maintains a priority on encouraging the conversion of nuclear reactor power sources from highly enriched to low enriched uranium. This EIS does not evaluate management alternatives for future receipts of domestic or foreign research reactor spent nuclear fuel. Rather, it evaluates the management of nuclear material (including spent nuclear fuel) already at the SRS. DOE evaluated many of the commentors' suggestions for the interim management of material that poses a safety or environmental concern. For example, DOE proposes to blend highly enriched uranium down to low enriched uranium. However, the scope of this EIS does not cover the evaluation of management alternatives for the long-term storage or disposition of spent fuel.

19-Apr-1995 01:37pm EST

To: Arthur B. Gould, Jr.

From: Robert H. Wilcox

Comments on Interim Management of Nuclear Materials Draft EIS

Thank you for sending me a copy of DOE/EIS-0220D, "Draft Environmental Impact Statement, Interim Management of Nuclear Materials", dated March 1995.

I have reviewed this document and wish to offer the following comments:

L5-1 1. The EIS Process. The management of nuclear materials, near-term, interim, and long term, is of course an important ongoing responsibility of the Department of Energy. This responsibility continues to require many day-to-day decisions in order to insure that the materials are appropriately and safely managed in a cost-effective way. In my opinion, it stretches the imagination beyond credibility to conclude that this should be considered a Major federal action in the context of NEPA and thus to undertake such a formal Environmental Impact Statement as this.

L5-2 2. The Draft EIS. The document itself appears to have been accurately and responsibly prepared. Those who prepared it and the DOE should be commended for doing a first-class job.

L5-3 3. Environmental Impacts. The differences in impacts among alternatives studied appear to have been accurately described. There is no reason to question the report's statement that "DOE expects the environmental impacts to be small for any of the scenarios..." Thus, once again, the NEPA process would not appear to be the appropriate way for DOE to arrive at the many decisions needed to manage this nuclear material.

L5-4 4. The Preferred Alternatives. The alternatives indicated as preferred for the listed materials in Table S-2 appear to be suitable and indeed the best in each case to get on with the task at hand in as expeditious a way as possible.

L5-5 5. Spent Fuel in the Receiving Basin for Offsite Fuels (RBOF). While admittedly of less urgency than dealing with the other material, it would be appropriate, in my opinion, to consider other alternatives for managing the research reactor fuel from RBOF. Various processing options should also be considered and the cost/benefit of each assessed. This need not be done as part of this EIS process, but DOE should be certain that the actions taken with regard to other materials leave open the option to process the research reactor fuel.

L5-6 6. Budgetary Considerations. There appears to be a very big disconnect between the DOE's preferred alternatives as stated in this document and recent decisions made by the Administration, including the DOE, concerning funding for the Savannah River Site (SRS). The SRS has

Comment L5. Page 1 of 2.

been asked to make do for these and other missions with far less money in FY96 than in FY95. For the budget in preparation for FY97, it has been reduced substantially beyond levels proposed for FY96. Outyears beyond FY97 have further reductions planned. Independently, the SRS was directed by DOE to reduce its staffing in FY95 by approximately 20%. Further, the DOE Financial Department has mandated sizable reductions in duly authorized FY95 and prior year project funds, which have forced cutbacks even in ongoing projects. When site missions are prioritized, it seems apparent that funds will be nowhere near sufficient to embark on the proposed alternatives described in this report. They will likewise be insufficient to comply with Recommendation 94-1 of the Defense Nuclear Facilities Safety Board. The DOE, in my opinion, should make very clear to the DNFSB, the Congress, and the public what effect such funding and staffing reductions will have on the management of the nuclear materials at SRS and other sites. L5-6

I hope that these comments will be helpful to DOE in this important matter.

Comment L5. Page 2 of 2.

Response to comment letter L5:

L5-1

DOE believes that the management of nuclear materials at the SRS for purposes other that those for which the facilities were designed (i.e., the production of weapons material) is a major Federal action in the context of the National Environmental Policy Act. Because many materials are no longer required for national defense, several management alternatives are available that were not viable in the past. Therefore, DOE believes that it should inform the public and the decisionmaker about the various alternatives and their potential environmental impacts prior to the selection of a management strategy.

L5-2

Comment noted.

L5-3

The NEPA process is not the only means by which DOE will reach decisions on the management of nuclear materials at the SRS. Other factors include resource availability, including adequately trained personnel, and budget considerations. The EIS describes the environmental impacts of the alternatives so the decisionmaker knows the potential impacts before making a decision.

L5-4

Comment noted.

L5-5

The Draft EIS categorized all aluminum-clad fuel stored in the Receiving Basin for Offsite Fuel as "stable" because there had been no observed degradation of this fuel. Since the publication of the Draft EIS, the Defense Nuclear Facilities Safety Board has stated that some of the material in RBOF has "significant corrosion," as discussed in Section 1.3. DOE placed fuel previously used in the Taiwan Research Reactor and slugs from the Experimental Breeder Reactor program in cans before placing that material in RBOF because of its poor condition. DOE knew the material was degraded before placing it in RBOF, and repackaged it as an interim measure. DOE believed this repackaging would be sufficient for interim storage until it could make disposition decisions. However, it now appears that some of the canisters in which the TRR fuel and EBR-II slugs was placed might be releasing radioactivity to the basin water because gas evolution has been observed from some canisters. While gas evolution does not prove the release of radioactivity to the basin water, it does suggest that the integrity of the particular canisters might not be intact. DOE does not believe that any substantial corrosion can be attributed to the RBOF water chemistry, which is still of high quality.

However, given the known poor condition of the material inside these canisters and the suspect condition of the canisters themselves, DOE has decided to categorize the TRR fuel and EBR-II slugs as candidates for stabilization. Chapter 2 now discusses applicable alternatives for the TRR fuel and EBR-II slugs, and Chapters 4 and 5 and Appendixes D and E now include data on this material group.

DOE is reviewing the long-term management of research reactor fuel in RBOF in other NEPA evaluations. The *Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs EIS* evaluates how DOE would manage this fuel if it implemented chemical processing technology. DOE recognizes that if its decision based on the *Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel EIS* implements a storage management technology, that EIS would not encompass the existing research reactor fuel in RBOF and an additional NEPA evaluation would be required. That evaluation would include all reasonable spent nuclear fuel management alternatives, including processing technologies.

L5-6

DOE and SRS budgets are being, and will continue to be, affected as Congress and the Administration restructure and prioritize the responsibilities and functions of the Department. DOE and SRS are aware of the commitments made in response to Defense Nuclear Facilities Safety Board Recommendation 94-1 and of the budgetary implications of those commitments. Budget realities will play an important role in the selection of management strategies.

State of South Carolina
Office of the Governor

DAVID M. BEASLEY
GOVERNOR

OFFICE OF EXECUTIVE
POLICY AND PROGRAMS

April 18, 1995

Mr. Arthur B. Gould, Jr.
Environmental Compliance Division SR NEPA
U.S. Department of Energy
Savannah River Operation Office
Post Office Box 5031
Aiken, South Carolina 29804-5031

Project Name: Draft Environmental Impact Statement Interim Management Of Nuclear Materials DOE/EIS-0220D

Project Number: EIS-950307-007

Dear Mr. Gould,

The Grant Services Unit, Office of the Governor, has conducted an intergovernmental review on the above referenced activity as provided by Presidential Executive Order 12372. All comments received as a result of the review are enclosed for your use.

The State Application Identifier number indicated above should be used in any future correspondence with this office. If you have any questions call me at (803) 734-0485.

Sincerely,

[redacted]

Rodney P. Grizzle
Grants Services Supervisor

Enclosures

Comment L6. Page 1 of 9.

Office of the Governor • Grant Services
South Carolina Project Notification and Review
1205 Pendleton Street
Room 477
Columbia, SC 29201

State Application Identifier
EIS-950307-007

Suspense Date
4/4/95

Mr. David Thompkins
South Carolina Department of Agriculture

The Grant Services Unit, Office of the Governor is authorized to operate the South Carolina Project Notification and Review System (SCPNRS). Through the system the appropriate state and local officials are given the opportunity to review, comment, and be involved in efforts to obtain and use federal assistance, and to assess the relationship of proposals to their plans and programs.

Please review the attached information, mindful of the impact it may have on your agency's goals and objectives. Document the results of your review in the space provided. Return your response to us by the suspense date indicated above. Your comments will be reviewed and utilized in making the official state recommendation concerning the project. The recommendation will be forwarded to the cognizant federal agency.

Should you have no comment, please return the form signed and dated.

If you have any questions, call me at (803) 734-0495.

RECEIVED
95
GRANT SERVICES

Rodney Grizzle

☐ Project is consistent with our goals and objectives.

☐ Request a conference to discuss comments.

☐ Please discontinue sending projects with this CFDA# to our office for review.

☐ Comments on proposed Application is as follows:

Signature: [redacted]

Date: 3-23-95

Title: Assistant Commissioner

Phone: 734-2210

Comment L6. Page 2 of 9.

Office of the Governor • Grant Services
South Carolina Project Notification and Review
1205 Pendleton Street
Room 477
Columbia, SC 29201

State Application Identifier EIS-950307-007
Suspense Date 4/4/95

Barbara Derrick
South Carolina Department of Labor

The Grant Services Unit, Office of the Governor is authorized to operate the South Carolina Project Notification and Review System (SCPNRS). Through the system the appropriate state and local officials are given the opportunity to review, comment, and be involved in efforts to obtain and use federal assistance, and to assess the relationship of proposals to their plans and programs.

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Should you have no comment, please return the form signed and dated.

If you have any questions, call me at (803) 734-0495.

RECEIVED
Rodney Grizzle
MAR 23 1995
GRANT SERVICES

☒ Project is consistent with our goals and objectives.

☐ Request a conference to discuss comments.

☐ Please discontinue sending projects with this CFDA# to our office for review.

☐ Comments on proposed Application is as follows:

Signature: [redacted] Date: 3/25/95

Title: Administrator Phone: 734-9637

Comment L6. Page 3 of 9.

Office of the Governor • Grant Services

South Carolina Project Notification and Review

1205 Pendleton Street
Room 477
Columbia, SC 29201

State Application Identifier
EIS-950307-007

Suspense Date
4/4/95

Olney England
Office Of Community Grant Program

MAR 22 1995

The Grant Services Unit, Office of the Governor is authorized to operate the South Carolina Project Notification and Review System (SCPNRS). Through the system the appropriate state and local officials are given the opportunity to review, comment, and be involved in efforts to obtain and use federal assistance, and to assess the relationship of proposals to their plans and programs.

Please review the attached information, mindful of the impact it may have on your agency's goals and objectives. Document the results of your review in the space provided. Return your response to us by the suspense date indicated above. Your comments will be reviewed and utilized in making the official state recommendation concerning the project. The recommendation will be forwarded to the cognizant federal agency.

Should you have no comment, please return the form signed and dated.

If you have any questions, call me at (803) 734-0495.

Rodney Grizzle

RECEIVED
MAR 28 1995
GRANT SERVICES

☐ Project is consistent with our goals and objectives.

☐ Request a conference to discuss comments.

☐ Please discontinue sending projects with this CFDA# to our office for review.

☐ Comments on proposed Application is as follows:

No Comment

[redacted]

Signature: Date: 3-27-95

Title: Phone:

Comment L6. Page 4 of 9.

Office of the Governor • Grant Services

South Carolina Project Notification and Review

1205 Pendleton Street
Room 477
Columbia, SC 29201

State Application Identifier
EIS-950307-007

Suspense Date
4/4/95

Earl F. Brown, Jr.
South Carolina Human Affairs Commission

The Grant Services Unit, Office of the Governor is authorized to operate the South Carolina Project Notification and Review System (SCPNRS). Through the system the appropriate state and local officials are given the opportunity to review, comment, and be involved in efforts to obtain and use federal assistance, and to assess the relationship of proposals to their plans and programs.

Please review the attached information, mindful of the impact it may have on your agency's goals and objectives. Document the results of your review in the space provided. Return your response to us by the suspense date indicated above. Your comments will be reviewed and utilized in making the official state recommendation concerning the project. The recommendation will be forwarded to the cognizant federal agency.

Should you have no comment, please return the form signed and dated.

If you have any questions, call me at (803) 734-0495.

Rodney Grizzle

RECEIVED
GRANT SERVICES

☑ Project is consistent with our goals and objectives.

☐ Request a conference to discuss comments.

☐ Please discontinue sending projects with this CFDA# to our office for review.

☐ Comments on proposed Application is as follows:

Signature: [redacted] Date: 3-28-95

Title: Executive Assistant, Human Affairs Phone: 253-6322

Comment L6. Page 5 of 9.

Office of the Governor • Grant Services
South Carolina Project Notification and Review
1205 Pendleton Street
Room 477
Columbia, SC 29201

State Application Identifier
EIS-950307-007

Suspense Date
4/4/95

Dr. James A. Timmerman, Jr.
South Carolina Wildlife and Marine Resources Department

The Grant Services Unit, Office of the Governor is authorized to operate the South Carolina Project Notification and Review System (SCPNRS). Through the system the appropriate state and local officials are given the opportunity to review, comment, and be involved in efforts to obtain and use federal assistance, and to assess the relationship of proposals to their plans and programs.

Please review the attached information, mindful of the impact it may have on your agency's goals and objectives. Document the results of your review in the space provided. Return your response to us by the suspense date indicated above. Your comments will be reviewed and utilized in making the official state recommendation concerning the project. The recommendation will be forwarded to the cognizant federal agency.

Should you have no comment, please return the form signed and dated.

If you have any questions, call me at (803) 734-0495.

Rodney Grizzle

GRANT SERVICES

☐ Project is consistent with our goals and objectives.

☐ Request a conference to discuss comments.

☐ Please discontinue sending projects with this CFDA# to our office for review.

☐ Comments on proposed Application is as follows:

Signature: [redacted] Date: 3/30/95

Title: Environmental Programs Div. Phone:

Comment L6. Page 6 of 9.

Office of the Governor • Grant Services
South Carolina Project Notification and Review
1205 Pendleton Street
Room 477
Columbia, SC 29201

State Application Identifier EIS-950307-007
Suspense Date 4/4/95

Steve Davis
S.C. Department of Health and Enviromental Control

The Grant Services Unit, Office of the Governor is authorized to operate the Soutl Carolina Project Notification and Review System (SCPNRS). Through the system the appropriate state and local officials are given the opportunity to review, comment, and be involved in efforts to obtain and use federal assistance, and to assess the relationship of proposals to their plans and programs.

Please review the attached information, mindful of the impact it may have on you agency's goals and objectives. Document the results of your review in the space provided. Return your response to us by the suspense date indicated above. Your comments will be reviewed and utilized in making the official state recommendatic concerning the project. The recommendation will be forwarded to the cognizant federal agency.

Should you have no comment, please return the form signed and dated.

If you have any questions, call me at (803) 734-0495.

Rodney Grizzle

RECEIVED
1995
GRANT SERVICES

☒ Project is consistent with our goals and objectives.

☐ Request a conference to discuss comments.

☐ Please discontinue sending projects with this CFDA# to our office for review.

☒ Comments on proposed Application is as follows:

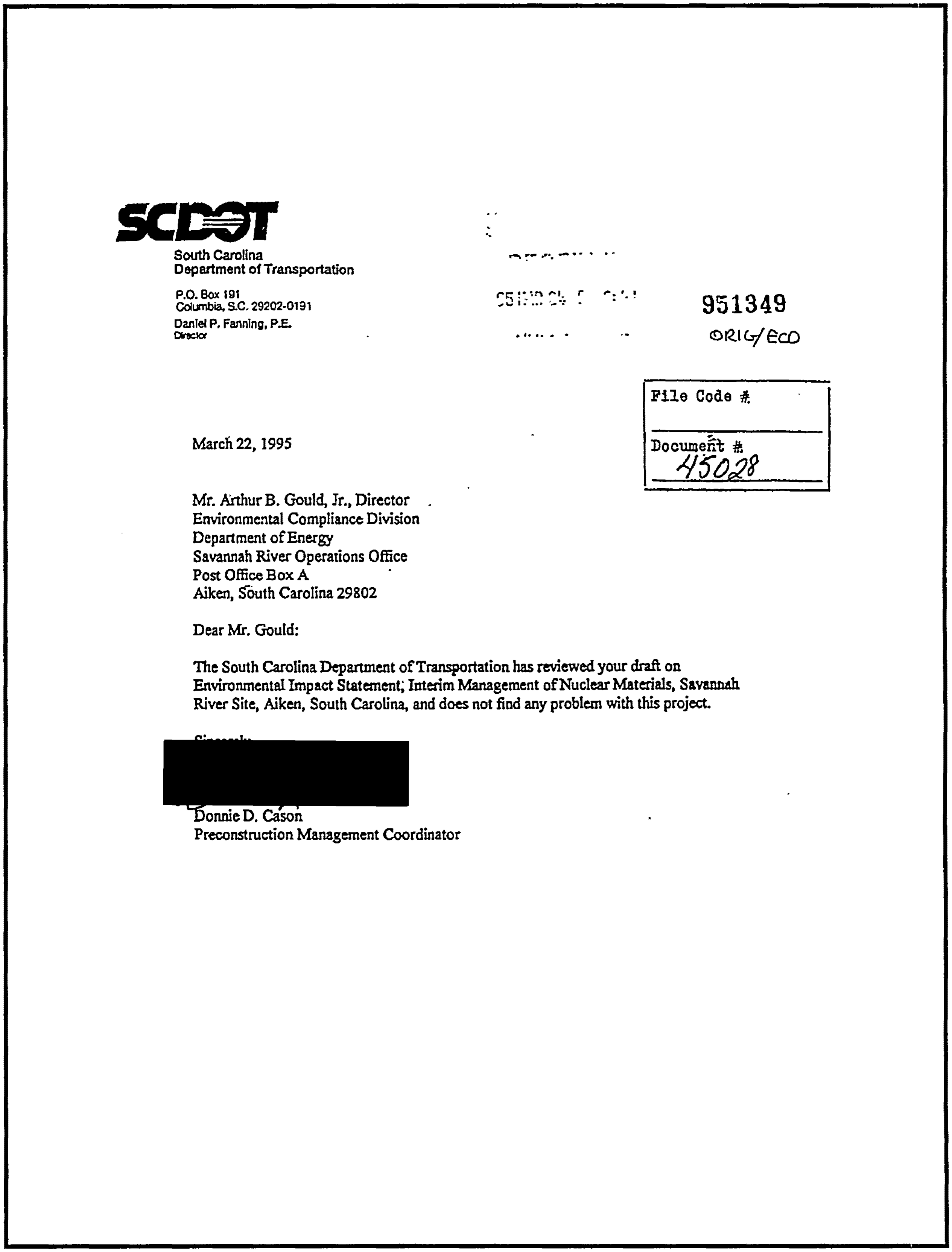
We are presently reviewing the document and will provide our comments directly to the U.S. Department of Energy.

Signature: [redacted]
Virgil R. Autry

Date: 4/3/95

Title: Director, Div. of Radioactive Waste

Phone: (803) 896-4244



SCDOT
South Carolina
Department of Transportation
P.O. Box 191
Columbia, S.C. 29202-0191
Daniel P. Fanning, P.E.
Director

951349
ORIG/ECO

File Code #
Document # 45028

March 22, 1995

Mr. Arthur B. Gould, Jr., Director
Environmental Compliance Division
Department of Energy
Savannah River Operations Office
Post Office Box A
Aiken, South Carolina 29802

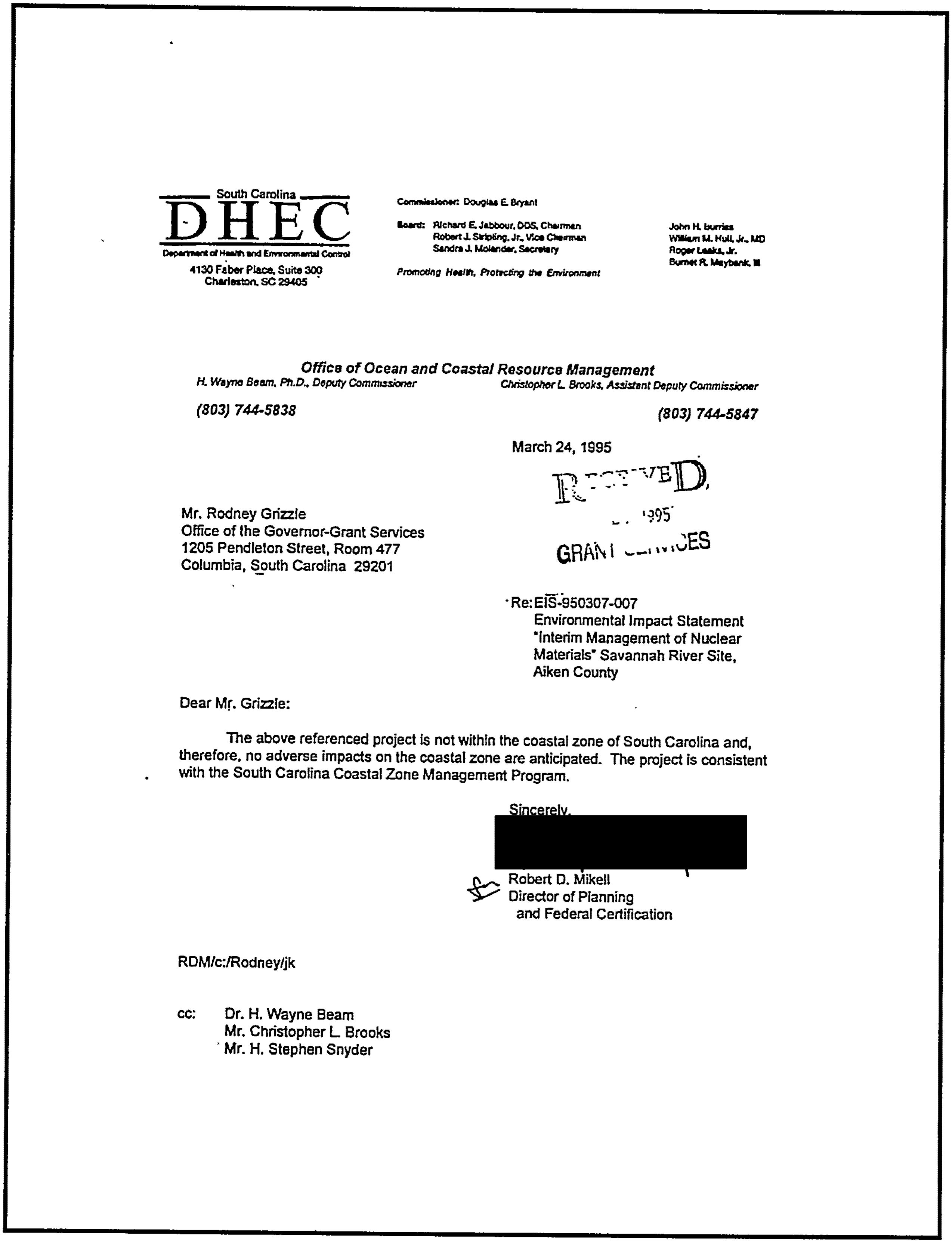
Dear Mr. Gould:

The South Carolina Department of Transportation has reviewed your draft on Environmental Impact Statement; Interim Management of Nuclear Materials, Savannah River Site, Aiken, South Carolina, and does not find any problem with this project.

[redacted]

Donnie D. Cason
Preconstruction Management Coordinator

Comment L6. Page 8 of 9.



South Carolina
DHEC
Department of Health and Environmental Control
4130 Faber Place, Suite 300
Charleston, SC 29405

Commissioner: Douglas E. Bryant

Promoting Health, Protecting the Environment

Office of Ocean and Coastal Resource Management
H. Wayne Beam, Ph.D., Deputy Commissioner — *Christopher L. Brooks, Assistant Deputy Commissioner*
(803) 744-5838 — *(803) 744-5847*

March 24, 1995

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[illegible] 1995
GRANT SERVICES

Mr. Rodney Grizzle
Office of the Governor-Grant Services
1205 Pendleton Street, Room 477
Columbia, South Carolina 29201

Re: EIS-950307-007
Environmental Impact Statement
"Interim Management of Nuclear
Materials" Savannah River Site,
Aiken County

Dear Mr. Grizzle:

The above referenced project is not within the coastal zone of South Carolina and, therefore, no adverse impacts on the coastal zone are anticipated. The project is consistent with the South Carolina Coastal Zone Management Program.

Sincerely,

[redacted]

Robert D. Mikell
Director of Planning
and Federal Certification

RDM/c:/Rodney/jk

cc: Dr. H. Wayne Beam
Mr. Christopher L. Brooks
Mr. H. Stephen Snyder

Comment L6. Page 9 of 9.

UNITED STATES ENVIRONMENTAL PROTECTION AGENCY
REGION IV
345 COURTLAND STREET. N.E.
ATLANTA. GEORGIA 30365

4PM-FA/mh

April 28, 1995

Mr. Arthur B. Gould, Jr., Director
Environmental Compliance Division
SR NEPA Compliance Officer
U.S. Department of Energy
Savannah River Operations Office
P.O. Box 5031
Aiken, South Carolina 29804-5031

SUBJECT: Draft Environmental Impact Statement (DEIS) on Interim Management of Nuclear Materials at SRS

Dear Mr. Gould:

The U.S. Environmental Protection Agency (EPA) has reviewed the subject DEIS in accordance with EPA's responsibilities under Section 309 of the Clean Air Act and Section 102(2)(C) of the National Environmental Policy Act (NEPA). The DEIS evaluates alternatives for stabilizing nuclear materials currently stored at SRS.

We are mostly concerned that DOE expedite the proposed actions to stabilize all liquid substances. Assuming that the DEIS conveys the correct status of these materials, and given the vulnerabilities documented by the Working Group Reports (Section 1.4.1), the need for action becomes increasingly critical as time goes by. In our view, the No Action alternative (continuing to store materials in current physical form) is unacceptable.

The Final EIS would benefit from a discussion of waste minimization (WM)/ pollution prevention (PP); e.g., how does the SRS WM and PP Awareness Plan apply to the proposed actions? Also consider adding the Pollution Prevention Act of 1990, and Executive Order 12856 (Federal Compliance with Right-to-Know and Pollution Prevention Requirements, August 1993) to the list of Laws and Requirements in Chapter 8. L7-1

Comment L7. Page 1 of 2.

Page Two

L7-2 The DEIS discusses in general the Programmatic Spent Nuclear Fuel (SNF) Management and INEL EIS (page 1-22 and Chapter 5). The FEIS should contain a more complete discussion in light of the recent issuance of the Final Programmatic EIS. The preferred alternative in that Programmatic EIS identifies SRS as the site to receive all DOE nationwide aluminum-clad SNF. Could this SNF be considered another candidate for stabilization?

L7-3 As stated on page 4-5, construction of a dry storage facility for SNF would be the the only alternative in this DEIS that could impact areas outside the industrialized F- and H-areas. The DEIS commits to the preparation of separate NEPA documentation if this activity is selected. We suggest that any such future NEPA documentation include a thorough discussion of cumulative impacts, especially given the extensive potential construction activities proposed in the recent SRS Waste Management EIS.

Based on our review, we rate this DEIS "EC-1." That is, while we have environmental concerns about the proposed actions, we believe the DEIS adequately sets forth the environmental impacts of the preferred alternatives. If you have any questions about our comments, please contact Marion Hopkins at 404/347-3776 (extension 6849).

Sincerely,

[redacted]

Heinz J. Mueller, Chief
Environmental Policy Section

Comment L7. Page 2 of 2.

Response to comment letter L7:

L7-1

Chapter 8 has been revised to incorporate a discussion of waste minimization and pollution prevention.

L7-2

The scope of this EIS is limited to materials currently at the SRS. The description of the Programmatic Spent Nuclear Fuel EIS in Section 1.6 has been revised to include the selected alternative, Regionalization by Fuel Type. The spent nuclear fuel data used in the cumulative impacts analysis in Chapter 5 represents regionalization by fuel type. The spent nuclear fuel that the SRS might receive could require stabilization to support extended (or long-term) storage.

L7-3

DOE has modified Chapters 4 and 5 to include estimates of environmental impacts from the construction of a Dry Storage Facility.

mpc
CHATHAM COUNTY-SAVANNAH METROPOLITAN PLANNING COMMISSION
2 E. BAY ST. BOX 1027 SAVANNAH, GA 31402, 912-236-9523

HOWARD BELLINGER, AICP
EXECUTIVE DIRECTOR

MILTON L. NEWTON, JR.
DEPUTY EXECUTIVE DIRECTOR

April 28, 1995

Mr. Arthur B. Gould, Jr.
NEPA Compliance Officer
US Department of Energy
PO Box 5031
Aiken, South Carolina 29804-5031

RE: "INTERIM MANAGEMENT OF NUCLEAR MATERIALS, SAVANNAH RIVER SITE, AIKEN, SOUTH CAROLINA" (DOE/EIS-0220D)

Mr. Gould:

I have reviewed the above referenced document and have several comments about it completeness.

NEPA requires all agencies of the Federal Government to prepare a detailed statement on the environmental effects of proposed major Federal actions that could significantly affect the quality of the human environment. The actions of the DOE and its contractors at the SRS in Aiken can and have significantly affected the quality of life in Savannah, Georgia.

L8-1 This impact statement fails to acknowledge that there is the potential for a spill of nuclear materials during proposed operations that could adversely impact the communities utilizing surface water from the Savannah River as their primary water source. The City of Savannah is being strongly encouraged by the Georgia Environmental Protection Division of the Georgia Department of Natural Resources to utilize the Savannah River more and more as the primary source of industrial and domestic water service. This statement does not address the possibility of a spill similar to the December 1991 tritium spill and the impact on communities as far away as Savannah, Georgia.

DOE can not ignore the possibility that a spill of nuclear
materials will contaminate the Savannah River at some future
time. The Clean Water Act prohibits discharge of toxic
L8-2 pollutants in toxic amounts to navigable waters of the US. The
Savannah River is a navigable water of the US. The Georgia EPD
must be involved with pollutant permit discharge decisions.

Comment L8. Page 1 of 2.

Therefore, the potential environmental impact of all operations at the SRS on human activities and their environment in Savannah, Georgia are within the scope of the EIS and should be fully addressed prior to the record of decision being issued. L8-3

Respectfully,

[redacted]

David A. Rutherford
Senior Environmental Planner

[redacted]

Harry Jue
Water & Sewer Director

cc: Michael B. Brown, City Manager
Dr. Mildred McClain, Citizens for Environmental Justice

Comment L8. Page 2 of 2.

Response to comment letter L8:

L8-1

The EIS acknowledges the potential for spill from nuclear facilities of radioactive material that could be released to the environment. Appendix E lists the accidents analyzed and presents the estimated consequences to the maximally exposed offsite individual (in terms of dose) and to the offsite public (in terms of potential latent cancer fatalities). In addition to estimated consequences from accidents that could release radioactive or hazardous material to the offsite environment, Chapter 4 presents impacts from liquid and airborne releases that are likely to occur from normal operations; these include regulatory permitted liquid discharges to onsite streams or creeks that flow into the Savannah River.

L8-2

DOE uses a number of outfalls to discharge liquid effluents from facility processes to SRS streams, creeks, lakes, and ponds. All of these outfalls are permitted under the National Pollution Discharge Elimination System (NPDES). None of the actions proposed in this EIS would exceed the limits in the NPDES permits or require the permitting of any additional outfalls (see Chapter 4).

L8-3

Chapter 5 presents the cumulative impacts to populations on and around the SRS, including downstream residents and water users in Savannah, Georgia, and its vicinity expected from normal operations that could result from actions proposed in this EIS. Chapter 5 also contains an estimate of the cumulative impacts that could occur from other actions DOE is considering that would involve the SRS. These include actions that would involve all SRS facilities.

633 Sandhurst Place
Aiken, S.C. 29801
April 30, 1995

Mr. Arthur B. Gould, Jr.
NEPA Compliance Officer
U.S. Department of Energy
P. O. Box 5031
Aiken, S.C. 29804-5031

Dear Mr. Gould:

This is a comment on the Draft Environmental Impact Statement on the Interim Management of Nuclear Materials, Savannah River Site, Aiken, S.C. (DOE/EIS-0220D).

I am pleased to say that from my standpoint, this Draft EIS is correct in the proposed handling and disposition of curium-containing residues (Cm-244 through Cm-248) now stored at SRS either as process solutions or as irradiated target assemblies.

I congratulate you for identifying these invaluable and irreplaceable materials which are now urgently needed for the continued production of Cf-252, particularly in the HFIR reactor at ORNL. I suggest that further information on the present uses of Cf-252 should be added. L9-1

Twenty-five years ago, I was working at the Savannah River Laboratory, under duPont's tenure at the Savannah River Plant. My job was to promote a particular neutron-emitting radioisotope, Californium-252.

Last week, as a duPont retiree, I attended a meeting at the Oak Ridge National Laboratory (ORNL) called the "Californium-252 Users' Workshop." This meeting attended by 89 scientitsts representing about 25 industrial and U.S. Government users of Cf-252, revealed that Cf-252 is alive and well, and that ORNL is capably handling the job of producing Cf-252 in their High Flux Irradiation Reactor (HFIR) and preparing this radioisotope for very practical and urgent industrial and military uses.

I also learned from Dr. C. W. Alexander of ORNL that there are two irreplaceable groups of HFIR feedstocks now available at the Savannah River Site. First, there are 65 target assemblies in RBOF containing a total of 762 grams of curium isotopes (Cm-244 through Cm-248) that represent a 20-year stockpile of feed for the HFIR at present loading rates. Second, in Tank 17.1 of the F-Canyon, there are americium and curium residues that represent 160 years stockpile for the HFIR. It is essential that the contents of Tank 17.1 be treated so that they can eventually be recovered at ORNL as feed for the production of Cf-252, and that the 65 target assemblies stored in RBOF be directed to ORNL as feed for Cf-252 production. L9-2

The reason for writing this letter to you is to point out that it would be a crime if these irreplaceable materials were converted to borosilicate glass and buried in a permanent repository. These curium isotopes, if used L9-3

Comment L9. Page 1 of 2.

L9-3 as feedstocks for the HFIR in the production of Cf-252, will assure the United States of a continuing supply of Cf-252, which is essential because of its relatively short half-life of 2.65 years. Production must continue to keep ahead of decay, much less to increase the number of uses that are now being proposed for Cf-252.

The following specific comments are made to various sections of DOE/EIS 0220D. My comments are from the standpoint of an individual who has not had direct contact with the program for many years. I suggest that you confirm any changes or additions to the EIS related to curium feedstocks for Cf-252 to Dr. Charles W. Alexander (615-574-7071) or Dr. John E. Bigelow (615-574-6926) Chemical Technology Division, Radiochemical Engineering Development Center, ORNL.

Section 2.2 Programmatic Material

L9-4 Page 2-3 (I suggest that the second group of missions should be expanded as follows - shown in underlined text:)

* Americium-243 and curium-244, which DOE would maintain as a national asset to support research in nuclear medicine, nuclear chemistry, solid-state chemistry, and nuclear physics. The higher isotopes of curium, (Cm-244 through Cm-248) are irreplaceable feedstocks for the production of californium-252, used as a neutron source for both military and industrial applications.

Section 2.2.2 Americium and Curium

Page 2-7

L9-5 About 14,000 liters (3,800 gallons) of americium and curium solution are stored in a single tank (17.1) in F-Canyon (Figure 2-2 shows F-Area with the F-Canyon Building in the center.) Americium and curium are feed materials in the DOE National Heavy Metal and Advanced Neutron Source Program that produces a large number of heavier transuranium elements, including californium-252. Californium-252 has a wide variety of medical, commercial, and defense-related uses, which include: cancer treatment, the on-line assay of coal to comply with DOE emission regulations, on-line quality control in cement manufacture, nuclear safeguards, and neutron radiography for the non-destructive inspection of major structures in military aircraft.

Note: I have been advised that ORNL has a report - now in editing - that summarizes recent developments in applications of californium-252. See Dr. Charles W. Alexander for additional information.

In conclusion, let me say that the Draft EIS presents the idea of Programmatic Materials very well, but that some elaboration on the production and uses of californium-252 is in order. Perhaps Dr. Alexander of ORNL will elicit some authoritative comments from present users on the need for the continuing supply of californium-252, therefore supporting the need for the DOE program on Programmatic Materials at the Savannah River Site.

Very truly yours,

[redacted]

Philip H. Permar

Phone: (803) 648-4987

Comment L9. Page 2 of 2.

Response to comment letter L9:

L9-1

DOE has revised Section 2.2.2 of the EIS in response to this and other comments to represent the uses of californium-252 more accurately.

L9-2

In response to this and similar comments, DOE has changed Tables S-1 and 1-1 and Appendix A to identify specifically the Mark-18 targets in the Receiving Basin for Offsite Fuel and 60 Mark-51 slugs in the P-Reactor disassembly basin. In addition, DOE has modified Tables S-2 and 2-1 to indicate its preferred alternative for the americium and curium solutions [Vitrification (F-Canyon)] and the targets and slugs (No Action).

L9-3

DOE does not propose to vitrify americium and curium materials for disposition in a deep geologic repository. The preferred alternative for the americium and curium liquids is Vitrification (F-Canyon) and storage in canisters for eventual shipment to Oak Ridge National Laboratory. For the solid materials (targets and slugs), the preferred alternative is to continue to store the materials at the SRS until the DOE Heavy Isotope Program can fund a campaign to separate the desired isotopes from the target and slug materials.

L9-4

DOE has modified Section 2.2 in response to this comment.

L9-5

DOE has modified Section 2.2.2 in response to this comment.

DEPARTMENT OF THE AIR FORCE
HEADQUARTERS SACRAMENTO AIR LOGISTICS CENTER (AFMC)
McCLELLAN AIR FORCE BASE, CALIFORNIA

1 May 95

MEMORANDUM FOR NEPA Compliance Officer
ATTENTION: Arthur B Gould, Jr.

FROM: SM-ALC/TI-1
5335 Price Ave
McClellan AFB CA 95652-2504

SUBJECT: Need For Californium-252

1 Comments on the Draft Environmental Impact Statement, Interim Management of Nuclear Materials, Savannah River Site, Aiken, South Carolina (DOE/EIS-0220D) were requested to be submitted to you by 1 May 95.

2. As a user of californium-252 (Cf-252), I am providing the following comments on need for a continued source of Cf-252:

a. Cf-252 is used here at McClellan Air Force Base in the Maneuverable Neutron Radiography System (MNRS) to perform industrial neutron radiography on intact aircraft components.

L10-1 b. Six sources containing up to 30 milligrams of Cf-252 each are used at the MNRS.

c. It is projected that two sources per year will be required to be shipped to us to maintain the necessary neutron flux to perform the nondestructive inspections.

d. Besides U.S. Air Force aircraft current being inspected, possible future workloads include U.S. Army Apache helicopters, U.S. Navy F-14s, Canadian F-18s, and large radar arrays.

3. For further information contact Jeff Ching, SM-ALC/TIRH, at (916) 643-1023.

[redacted]

JEFFREY K.K. CHING
Acting Chief, Nuclear Licensing and Operations

cc:
Chuck Alexander, ORNL Radiochemical Engineering

Comment L10.

Response to comment letter L10:

L10-1

Section 2.2.2 of the EIS has been revised in response to this and other comments to represent more accurately the uses of californium-252. DOE recognizes the use of californium-252 sources in neutron radiography for the testing of aircraft components.

NRDC

Natural Resources Defense Council

1350 New York Ave., N.W.
Washington, DC 20005
202 783-7800
Fax 202 783-5917

May 1, 1995

Mr. A.B. Gould, Jr.
NEPA Compliance Officer
U.S. Department of Energy
Savannah River Operations Office
P.O. Box 5031
Aiken, SC 29804-5031

Attention: Interim Management of Nuclear Materials EIS

Dear Mr. Gould:

The Natural Resources Defense Council and the Energy Research Foundation submit the following comments on the Department of Energy's ("the Department's") Draft Environmental Impact Statement on Interim Management of Nuclear Materials at the Savannah River Site ("the EIS").

The EIS falls considerably short of the National Environmental Policy Act's ("NEPA's") requirements. It offers an illogical and unsupported set of decisionmaking criteria that would bias the agency decision in favor of processing in F- and H-Canyons ("the canyons"). It presents an inadequate range of alternatives and fails to discuss adequately some of the alternatives it does present. With unfortunate consistency, it fails to give the decisionmaker or the public the information they need on important issues.

No draft EIS is perfect, but these are major flaws They are so serious that the Department must address them and resubmit the EIS in draft form rather than proceed directly to a final document.

I. Illogical and Unsupported Decisionmaking Criteria.

Processing activities in the canyons result in two products: separated nuclear material and large amounts of liquid radioactive waste. It is a substantial understatement to observe that the United States hardly needs more of either of these products. Given these systemic problems with processing (to say nothing about the safety and cost problems of running the canyons), we had hoped that the Department would use this EIS to look critically at processing activities and to seriously evaluate other options. To our disappointment that has not

[illegible]

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Comment L11. Page 1 of 10.

happened, and the Department's decisionmaking criteria appear largely at fault.

The EIS makes two broad judgments in deciding how to manage the materials at issue. Neither judgment is supported, and together they heavily bias the preferred alternatives in favor of processing. The first judgment is to conclude that "there were no significant differences in environmental impacts among the alternatives." EIS at 2-13. This judgment is flatly wrong. For example, Table 2-9 demonstrates that for the Mark-31 targets there would be 6.5 cancer deaths[1] from an accident in the Processing to Metal Alternative, but less than one one-hundredth of a cancer death from an accident under the Improving Storage Alternative. Similarly, Table 2-9 reveals that there would be nearly twice as much high-level liquid radioactive waste generated by the Processing to Metal Alternative (2.1 million liters) than the Improving Storage Alternative (1.2 million liters). Gaps between these two alternatives in other categories, such as health effects of normal operations, are similarly large. L11-1

The EIS's second critical judgment further conspires against the selection of an alternative other than processing, and it does so systematically. Under this key decisionmaking criterion, the Department simply selected as preferred the alternative that could be implemented most quickly. EIS at 2-13. Elevating speed of implementation to the sole decisionmaking criterion is senseless from the standpoint of environmental impact and economic cost; by using this criterion as the dispositive one, the EIS repeatedly selects treatment options that have unnecessary environmental impacts and that almost certainly cost much more money than other alternatives. Just as importantly, it games the system squarely in favor of processing. Processing has been going on at Savannah River for more than 40 years, and as a familiar operation it can be implemented more quickly than other, more innovative alternatives. Thus, the EIS repeatedly selects processing as the preferred alternative. L11-2

We do not believe that speed of implementation is an irrelevant decisionmaking criterion. For those materials that present real and immediate safety problems, it may well make sense to choose the option that can be implemented most quickly. But even from the EIS's inadequate discussion of the safety risks presented by the categories of materials at issue, see EIS at 1-10 to 1-15, it is clear that some materials present safety concerns that are far from immediate. For these materials, a

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[1] We question below the EIS's methodology for estimating incidences of cancer.

L11-2 (cont.)

delay in implementing an alternative carries little cost and may present large benefits if by taking some additional time the Department can address the materials less expensively and with less environmental impact.

The Department should completely overhaul its decisionmaking criteria. First, it should acknowledge that some alternatives are better for the environment than others, and it should discuss those differences squarely in the EIS. Second, it should withdraw speed of implementation as the dispositive decisionmaking criterion and instead consider it as one of several factors -- along with environmental impact, economic cost, immediacy and scale of the safety concern from the materials in their current state, and other important factors -- in selecting its preferred alternatives.[2]

L11-3

II. <u>Inadequate Range of Alternatives.</u>

The Council on Environmental Quality's regulations for implementing NEPA call the discussion of alternatives "the heart of the environmental impact statement." 40 C.F.R. § 1502.14. An EIS must discuss a reasonable range of alternatives, and an EIS that fails to do so violates NEPA. <u>See</u>, <u>e.g.</u>, <u>Natural Resources Defense Council v. Morton</u>, 458 F.2d 827 (D.C. Cir. 1972). Because it contains an inadequate range of alternatives in at least two respects, this EIS violates NEPA. Its inadequate discussion of some alternatives also violates NEPA.

L11-4

A. <u>Failure to Include "Minimal Processing Alternatives Scenario."</u>

In the all-important Chapter 4, on "Environmental Impacts," the Department presents three groups of alternatives and compares and contrasts their effects in an attempt to meet NEPA's requirement for "sharply defining the issues and providing a clear basis for choice among options by the decisionmaker and the public." 40 C.F.R. § 1502.14. The first alternative presented is the no-action alternative, required by law to be included in the EIS. The second is the Department's preferred alternative, which is heavily weighted toward processing. The third, the "comparative alternatives scenario," is simply a collection of the alternative for each individual category of materials which has the greatest environmental impact. It is a

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[2] Because of the serious Resource Conservation and Recovery Act compliance problems the Department faces in operating the canyons to treat the materials in this EIS, we believe that the EIS consistently underestimates the time that would be necessary to implement processing alternatives involving the canyons.

straw man of the highest order -- because it purposely collects the worst of the worst in terms of environmental impact, the Department would never select it. The Department also would never select the no-action alternative, given its oft-expressed view that the materials at issue in the EIS require some form of treatment in order to be stabilized. Thus, in its all-important section comparing and contrasting the environmental impacts of the available alternatives, the EIS considers only one alternative that it could actually select: its preferred alternative, which features reprocessing. This is inadequate under the law. L11-4 (cont.)

To satisfy NEPA's requirements, the EIS must consider at least one "alternatives scenario" that improves the current storage situation for potentially vulnerable materials but seeks to do so without processing the materials in a manner that separates out fissile material. Only by seeing such a "minimal processing" alternative will the decisionmaker and the public be able to evaluate adequately the true options available for these materials.

B. <u>Failure to Consider Conversion to Glass Alternative for Plutonium-242 Solutions.</u>

The EIS's second alternatives-related inadequacy relates to the so-called "programmatic need" materials, the materials for which the Department sees an ongoing programmatic requirement. For at least one category of these materials, the plutonium-242 solutions, the Department unreasonably constrains the range of alternatives it considers based on its desire to separate and use the material programmatically. While the Department admits that conversion of the plutonium-242 solutions to glass is a viable alternative for "stabilizing" the material,[3] it fails to analyze and discuss this alternative in the EIS because the alternative would not place the Pu-242 in a form suitable for programmatic use. EIS at 2-7. But since conversion to glass would not <u>exclude</u> future programmatic use of the material if the Department decides, after completion of the Stockpile Stewardship and Management Programmatic Environmental Impact Statement, to go forward with hydrotests involving L11-5

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[3] We question below whether the plutonium-242 solutions really do need "stabilization" in light of the fact that they do not present a criticality danger, as the Department acknowledges.

plutonium, it is a reasonable alternative that must be evaluated in this EIS.[4]

C. Inadequate Discussion of Alternatives.

L11-6 Finally, the EIS threatens to bias the decisionmaker's ultimate selection of alternatives by failing to discuss fully and fairly some of the alternatives that it does present. For example, in its discussion of the preferred Processing to Oxide Alternative for the plutonium-242 solutions, the Department fails to discuss or acknowledge in any way the substantial dispersability and leachability problems posed by converting this material to oxide form. In a similar vein, the Improving Storage Alternative for the Mark-31 targets fails to discuss actions the Department could take prior to construction of a dry storage facility in order to enhance the current storage situation -- improving the deionization capability in existing storage basins, for example. These are serious omissions, and they render the EIS's discussion of alternatives inadequate.

III. Inadequate Discussion and Analysis of Important Issues.

The EIS consistently fails to contain sufficient discussion and analysis to give the decisionmaker and the public the information they need and are entitled to under NEPA.

A. Failure to Disclose Underlying Data and Assumptions.

L11-7 Repeatedly, the EIS presents conclusions for which it fails to provide the data and assumptions that went into deriving them. For example, Table 2-9 presents the important conclusion that operating the canyons to process the Mark-31 targets into metal would result in 0.00025 population latent cancer fatalities, but nowhere does the EIS reveal the data, assumptions, and models that went into arriving at this conclusion. There is thus no way for the public to evaluate the adequacy of the Department's methodology and the accuracy of its conclusion. To comply with NEPA's requirement for informed public participation, the Department must reveal all of the relevant data, assumptions, models, and calculations that went into its discussion and analysis of issues treated in the EIS.

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[4] The Department's proposed hydrotesting activities fall within the scope of the Stockpile Stewardship and Management Programmatic EIS ("PEIS"). Under 40 C.F.R. § 1506.1(c) the Department cannot take an environmentally damaging action that is part of this program if it would prejudice the Department's ultimate decision on the PEIS, as making separated Pu-242 readily available would.

Comment L11. Page 5 of 10.

L11-8 Many of the information omissions are basic and egregious -- for example, failing to reveal the quantity of fissile material that would be isolated from processing the substances at issue.[5]

L11-9 The Department must do a particularly good job in explaining and justifying the assumptions that went into its estimates of the time required to perform certain specific tasks necessary to implement alternatives. For example, for the H-Canyon plutonium-239 solutions, the Processing and Storage for Vitrification Alternative claims that it would take a full three years to transfer the solution from H-Canyon to the H-Area high-level waste tanks. EIS at 2-15. Even more implausibly, the EIS claims that it would take more than ten years to implement a dry storage alternative for the Mark-31 targets.[6] EIS at 2-30.

B. <u>Inadequate Discussion of Nonproliferation Issues.</u>

The EIS's discussion of the proposed action's nuclear nonproliferation impacts occupies only four short paragraphs on page 2-46. This discussion is patently inadequate in light of the important consequences this proposed action would have for nonproliferation interests.

L11-10 After having its chemical separation activities shut down for three years (with the only exception a limited clean-out run of solutions in F-Canyon), the Department is now proposing to reenter the processing business in a substantial way. Such a decision would have crucial implications for other countries' activities. Most importantly, the ability of the United States to persuade Russia to suspend its separation of plutonium would be seriously undercut by a U.S. return to large-scale chemical separation activities. The EIS entirely ignores these crucial impacts, which violates NEPA.[7] The document's failure to

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[5] In light of Secretary O'Leary's Openness Initiative, and especially in light of her release of data on most fissile material inventories, there is no legitimate classification-based justification for not releasing this information.

[6] The EIS's selection of a ten-year window in which an alternative must be able to be implemented is entirely arbitrary and is supported by no evidence.

[7] Moreover, the agency decision here is not being made in a vacuum, as other proposals for processing activities are under consideration by the Department in venues such as the Foreign Research Reactor EIS. The potential for the agency action here to serve as the beginning of a larger campaign of processing for waste management purposes, which would have dramatic nuclear

discuss the nonproliferation implications of separating plutonium-242 for use in hydrotests, which is likely to encourage similar behavior by other nations, similarly violates NEPA.

What little nonproliferation discussion appears in the EIS is badly inadequate and in some ways misleading. For example, the Department asserts that it has committed not to use for nuclear explosive purposes Pu-239 or highly enriched uranium separated as a result of this agency action, but it fails to reveal that the separated material will nevertheless be weapon-grade and -useable, that the commitment could be reversed by a future Administration, and that the commitment does not apply to the Pu-242 at issue in the EIS.

C. Failure to Discuss Economic and Budgetary Issues.

L11-11 While the Department acknowledges on page 2-45 the importance of budgetary and economic considerations to the decision at issue here, the EIS fails to discuss these critical issues. In addition to violating NEPA, the absence of such an economic discussion is very poor public policy. The canyons are tremendously expensive to run, and the cost advantages of choosing non-processing alternatives demand a full analysis. This is particularly true in light of the need for costly upgrades in the canyons' exhaust, fire protection, seismic, and other systems, all of which must be discussed in the EIS. In a time of extreme budget scarcity, budgetary tradeoffs between running the canyons and cleaning up the weapons complex need full disclosure and analysis to the public and high-level agency decisionmakers.

On a related matter, the EIS should contain a discussion of staging alternatives -- that is, options for compressing the operations of the canyons if some processing is ultimately selected, in order to reduce the costs and environmental impacts of operating the facilities.

D. Inadequate Discussion of Programmatic Need.

1. In general.

L11-12 The EIS assumes rather than assesses the existence of a continuing programmatic need for the materials at issue. The absence of such a discussion violates NEPA, because programmatic requirements form part of the asserted purpose and need for the

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proliferation consequences, underlines the need for a comprehensive, prospective analysis in this EIS concerning the nonproliferation implications of processing.

Comment L11. Page 7 of 10.

agency action here. As part of this discussion, the EIS should analyze the very high per-unit costs of obtaining materials such as plutonium-242, americium-243, curium-244, and neptunium-237 by running the canyons. L11-12

2. Plutonium-242.

The case of the plutonium-242 is important enough to warrant its own discussion. The "programmatic need" for which the Department desires to use Pu-242 is hydrotesting. The mere fact that such testing involves non-fissile isotopes of plutonium is currently classified, which makes little sense for at least two reasons. First, the fact of the testing program is an open secret that has been covered in the media. Second, such testing has crucial implications for nuclear nonproliferation and arms control interests generally, and for a comprehensive nuclear test ban specifically. As such, it demands a full and public discussion, not banishment to a shadowy classified appendix. We specifically do not seek information related to shapes or configurations, or other specific data that the Department should keep secret for nonproliferation reasons. There is no legitimate cause, though, to keep secret the fact and broad outlines of the hydrotesting program, which demand an intelligent policy debate. We specifically request that the Department undertake an expedited classification review of this program. L11-13

E. Inadequate Discussion of Safety Risks Posed By Materials In Their Present Form.

The central purpose of the proposed agency action is to "stabilize" materials the Department believes to present a safety concern. Given the fact that this concern about safety is the driving force behind the proposed action, the discussion at pages 1-10 through 1-15 concerning safety risks posed by the status quo is conclusory, unanalytic, and in some cases seemingly illogical. L11-14 For example, the one-sentence description of the safety situation concerning the fuel assemblies in the H-Canyon Storage Basin is that "[n]o corrosion has been detected." EIS at 1-12. The Department nevertheless deems these materials candidates for stabilization because they are similar to other materials and are stored in a similar environment. Id. L11-15

As another example, the EIS acknowledges that the plutonium-242 solutions and some of the americium and curium solutions do not present a criticality hazard, but it nevertheless concludes that "there is a need to stabilize these solution independent of the program need." EIS at 2-3 to 2-4. This is a non-sequitur, and the safety situation posed by these and the rest of the materials demand real and substantive analysis in the EIS. L11-16

F. Inadequate Discussion of Safety of Canyon Operation.

L11-17 While Appendix E contains a variety of safety information, the body of the EIS (i.e., the part of the document that the general public actually reads and can understand) is largely bereft of information about whether the canyons are safe in the event of an accident. The section on page 2-48 concerning "Aging Facilities" fails to discuss in a substantive way the age-based safety problems posed by canyon operation. In light of the 1992 shutdown of the canyon due to concerns about the seismic integrity of the canyons' ventilation system and given the susceptibility of the area to earthquakes, see EIS at 3-7, the EIS must in particular contain a full and direct discussion of the safety of the canyon ventilation systems. The EIS should also describe the waivers, compensatory measures, and other similar procedures under which the canyons would operate.

G. Understating Latent Cancer Risk

L11-18 The Department assumes that the latent cancer risk is 0.0005 fatalities per person-rem for members of the public and 0.0004 fatalities per person-rem for workers, but it fails to explain how these risk factors were derived. We assume they are based on Table 4.2 of National Research Council, Health Effects of Exposure to Low Levels of Ionizing Radiation (1990) ("BEIR V"), with a dose-reduction factor of 2 for low dose rates.

These cancer risk estimates should be increased by a factor of 2 due to the lack of scientific support for the dose rate reduction factor in human data, see BEIR V at 22; by a factor of 1.5 to 2 in order to account for risk in terms of cancer incidence rather than just in terms of cancer fatalities, see National Research Council, The Effects on Populations of Exposure to Low Levels of Ionizing Radiation: 1980 180, 214 (1980) ("BEIR III"); and by a factor of 1.25 to account for the cancer risk to individuals who will get cancer from other causes, see BEIR V at 174, for a total increase by a factor of 3.75 to 5. Thus, the appropriate latent cancer risk is 0.0015 to 0.002 cancers per person-rem for workers and 0.001875 to 0.0025 cancers per person-rem for members of the public. The Department must revise the EIS's analysis of radiological health effects to comport with these scientifically appropriate risk estimates.

H. Inadequate Discussion of Compliance With Other Laws.

L11-19 The EIS fails to present an adequate discussion of the proposed action's compliance with legal requirements other than NEPA. The substantial air emissions from operation of the canyons raise basic questions about the facilities' compliance with the Clean Air Act's provisions for hazardous air pollutants,

Comment L11. Page 9 of 10.

questions that the EIS does nothing to answer in its discussion on pages 8-1 to 8-2. L11-19 (cont.) In addition, the Department appears to be out of compliance with the Resource Conservation and Recovery Act ("RCRA") for its proposed operation of the canyons, and the vague discussion in the EIS fails to reveal whether and how the Department will undertake compliance with RCRA. See EIS at 8-2 to 8-3. L11-20

I. Inadequate Discussion of Air Impacts and Waste Generation.

In addition to nonproliferation and safety impacts, the most significant effects from the proposed processing alternatives appear to be in the areas of air emissions and waste generation. The EIS discusses neither of these consequences adequately: The discussion on air impacts fails to highlight in layperson's language the substantially higher radioactive air emissions from the processing alternatives as compared to other alternatives; understates the risk of cancer from these air emissions; and includes a sentence (the first full sentence on page 4-31) that makes no sense. L11-21

The EIS's substantive discussion on waste generation consists of a single unanalyzed and undeveloped table that presents amounts of waste to be generated, followed by a single sentence proclaiming that the impacts of this waste generation would be "minimal." EIS at 4-40. The EIS needs to explain in detail why this is so. In light of the Department's checkered history in the waste management area and in light of the substantial amount of different types of waste that would be generated by processing in the canyons, the Department must explain and justify in detail its conclusion that waste impacts would be insubstantial. L11-22

Sincerely,

[redacted]

Andrew P. Caputo
Attorney for the Natural Resources
Defense Council and the Energy
Research Foundation

cc: Secretary O'Leary
Under Secretary Curtis
Assistant Secretary Grumbly
Assistant Secretary O'Toole
General Counsel Nordhaus

Comment L11. Page 10 of 10.

Response to comment letter L11:

L11-1

DOE has deleted the language cited by the commentor. It was not DOE's intent to imply that there were no discernible differences in an absolute sense. Similarly, it was not DOE's intent to imply there was no difference between any of the alternatives for a single environmental factor. To enable better comparisons of the relative impacts between the alternatives, DOE has added several tables in Chapter 4 that provide a direct comparison among the alternatives for every environmental factor evaluated. The addition of these tables provides the readers the information with which they can draw their own conclusions on the relative magnitude or significance of impacts between alternatives. Further, DOE has included Section 2.7 to explain the reasons DOE selected the preferred alternative for each category of nuclear material.

L11-2

DOE has included additional information in Chapter 2 to clarify that the relative speed by which the Department could implement an alternative was only one of the factors or criteria for identifying that alternative as "preferred." As Chapter 2 indicates, DOE considered and evaluated each alternative against a number of criteria, including environmental and institutional factors. These criteria included (1) the availability of existing facilities or equipment with which to implement actions required by the alternative, (2) the degree to which an alternative would require the development of new technology (or advances in existing technology) for successful implementation (i.e., technological risk), (3) the estimated duration of activities required in the implementation of the alternative (including technology development, construction of facilities, and conversion or stabilization actions), and (4) the relative differences in environmental impacts among the alternatives. In addition, DOE acknowledges that it must weigh key institutional factors in arriving at its ultimate decision of which alternatives to implement. These factors include cost and budget considerations, impacts on national security and the U.S. policy on nonproliferation, the availability of trained technical personnel with whom to implement the alternatives, and the reliance on aging nuclear facilities versus the construction of new facilities required to implement the various alternatives.

The commentor points out that expediency of action should be closely related to the immediacy of the threat posed by a material to the environment or to the safety and health of the workers or the public. Technical experts in DOE and the Defense Nuclear Facility Safety Board (DNFSB) have characterized the materials and storage conditions as environmental, safety and health vulnerabilities that "should not be allowed to persist unremediated (DNFSB 1994)." The EIS analyzes alternatives to alleviate environmental, safety, and health concerns about nuclear material storage conditions at SRS.

The EIS acknowledges that, regardless of the alternatives selected for implementation, it could take nearly 10 years to complete all of the actions required. Nevertheless, DOE does believe that expediency of action is an important consideration because it has corollary implications related to cost and safety. For example, Figures 2-8 and 2-9 illustrate that by stabilizing nuclear materials, DOE could reduce future operating costs as it consolidated materials for storage in fewer facilities.

L11-3

The commentor apparently has interpreted the three scenarios presented in Chapter 4 of the Draft EIS as the only alternatives that DOE evaluated. This interpretation is not correct. DOE evaluated a range of reasonable alternatives for each of the 10 material categories (see Chapter 2). The number of possible combinations of these alternatives is extremely large. Therefore, DOE created the scenarios to represent a range of impacts that could result from the implementation of alternatives for each material category. To provide a clearer presentation and assure the reader that the Department analyzed a range of alternatives for each material, DOE has revised Chapter 4, including tables that summarize 10-year data to indicate the relative degree of impacts among the alternatives for each environmental factor. Further, DOE has moved the discussion of the scenarios to Chapter 5, which discusses cumulative impacts. The scenarios in Chapter 5, when included with potential impacts from other current and reasonably foreseeable activities that could have environmental implications at the SRS, represent the potential cumulative effects of actions proposed by the EIS. Thus, Chapter 4 now presents the impacts that could result from any of the alternatives evaluated by the EIS on a material-by-material basis, and Chapter 5 presents the range of potential cumulative impacts from the combination of alternatives and from other current and reasonably foreseeable actions at the SRS.

L11-4

At the suggestion of the commentor, DOE included a Minimum Processing Scenario in the EIS in Chapter 5. The Draft EIS contained the data for this and any other scenario a reader might wish to construct from a combination of the alternatives analyzed for each material.

L11-5

The Draft EIS contains an alternative for the vitrification of plutonium-242 by transporting the material, either in an oxide or solution form, from H-Area to F-Canyon (see page 2-6 of the Draft EIS). The alternative would require the modification of a portion of the F-Canyon to provide the vitrification capability and either the operation of the HB-Line to produce an oxide for transport or the development or acquisition of a container suitable for shipping the plutonium in a liquid form.

The Draft EIS did not evaluate vitrification of plutonium-242 at the Defense Waste Processing Facility (DWPF) because this alternative would not satisfy the purpose and need for action and because it would not preserve the plutonium-242 in a form usable for potential programmatic purposes. Vitrification in the DWPF would render plutonium-242 unusable by mixing it with the other constituents in the high-level waste system. This process would result in a glass composite with extremely high radiation levels and a significant number of impurities and other radioisotopes, and would dilute the plutonium-242 to the point where it would not be technically or economically reasonable for DOE to recover in the future. Despite the fact that this alternative is "unreasonable" for purposes of the National Environmental Policy Act review in this EIS, DOE has included it in this Final EIS in response to the commentor's suggestion (see Section 2.2.1).

L11-6

As the commentor points out, the materials pose relative degrees of hazard, depending on their physical form and the manner in which they are packaged and stored. DOE has added information to Appendixes A and B on the relative hazards posed by the various types and forms of nuclear materials evaluated in the EIS. The commentor specifically mentions the lack of discussion on the hazards posed by plutonium oxide. DOE acknowledges that plutonium, in any physical form, constitutes a significant radiological hazard and, therefore, has added the description of a potential Actinide Packaging and Storage Facility/Consolidated Vault in Appendix C that explains the hazards posed by plutonium oxide (or metal) and how the operations and packaging of the material in accordance with the DOE standard for plutonium storage would minimize these hazards.

The commentor also requests information about temporary or compensatory measures that DOE could take to alleviate a safety concern or vulnerability, such as improving basin water chemistry until it can develop a dry storage capability. DOE can and does take such compensatory actions; in fact, they are ongoing in many cases. DOE has expanded the discussion of the No-Action Alternative to describe these activities. However, these compensatory measures can only postpone the need to stabilize the material and alleviate the vulnerability or concern identified. For example, an increased deionization capability would not alter the irreversible corrosion of the fuel and targets stored in the water-filled basins; it simply would remove the radioactivity being released at a faster rate and improve the chemical quality (i.e., conductivity) of the basin water in an effort to maintain acceptable radiation levels in the immediate vicinity of the basin and to slow corrosion rates. This would not alleviate the root cause of the problem (i.e., materials stored under conditions for which they were neither designed nor intended), it simply would retard the effects.

L11-7

Tables such as Table 2-9 summarize the information in Chapter 4, which identifies impacts that could result from the implementation of the alternatives. Each section in Chapter 4 identifies the models, conversion factors, and many of the assumptions used to develop the estimates of environmental impacts. Appendix E contains similar information about the methods used to analyze impacts from postulated accidents. All of the models used to develop data for the EIS are widely accepted and used in the scientific community. DOE has used the models consistently in the development of other environmental studies, both Department-wide and at SRS. The EIS references all of the information and assumptions used to develop the models; this information is publicly available in DOE reading rooms.

L11-8

Information has been added to Appendix A on the total amount of weapons-grade plutonium produced and currently stored at the SRS. Information on the total metric tons of heavy metal contained in irradiated reactor fuels and targets is also presented. The exact quantities of special nuclear materials or isotopes stored in single containers or locations are classified because of sensitivities associated with theft, diversion, or sabotage. DOE performed a number of classification reviews to determine the extent of information that it could include in the EIS.

L11-9

DOE added new information in Chapter 2 to give the reader a better understanding of the basis for the implementation schedules shown, including dry storage for the Mark-31 targets. In general, the schedules were constructed using the following assumptions:

1. If the physical capability to implement the alternative currently exists, the estimated time to process, convert, or repackage the material was based on the historic throughput capacity of the particular facility coupled with recent operating experience in similar facilities (e.g., HB-Line). In all cases, an effort was made to provide a conservative estimate of the time required. This was not done to make alternatives appear to take a long time, but rather to ensure that any resulting environmental impacts were adequately estimated.

2. If the physical capability does not exist, engineering studies were used to estimate the time required to design and construct the necessary capability, and to estimate the capacity or throughput capability that any new facility (or modified existing facility) would have. These studies can be characterized as preliminary conceptual design reports or, in some cases, feasibility studies. Again, similar to the case of existing facilities, DOE believes the time estimates associated with the design, construction, and operation of any new capabilities are

conservative. DOE included any time required by law to obtain Congressional approval and authorization where the scope of a project or new facility so requires (e.g., a line item project in the Federal budget to construct a new facility for the dry storage of irradiated reactor fuel or targets). DOE also included the time to implement the requirements of the Department's acquisition process for a project of this magnitude.

3. To implement several alternatives using existing facilities or capabilities, some level of research and development would be required. This is particularly true for alternatives that would involve the use of existing SRS high-level waste processing facilities (including the Defense Waste Processing Facility) to vitrify nuclear materials. These alternatives would require extensive analytical laboratory research to develop new chemical processes or to validate that existing processes could be used without extensive equipment or facility modifications. The engineering studies used to evaluate these alternatives were essentially feasibility studies. The technical complexity that would be associated with implementing these alternatives is such that DOE conservatively believes it would take several years to demonstrate their technical feasibility even on a laboratory scale. For example, to send plutonium-239 to the waste tanks for subsequent vitrification in DWPF, it would be essential to understand, through laboratory testing, the potential for this material to precipitate or concentrate in each part of the process to ensure that no criticality event would occur. It would also be necessary to ensure an understanding of where the plutonium would be distributed in the high-level waste before it was subjected to the in-tank precipitation (ITP) and feed preparation processes before introduction into the DWPF. Because the complexity of the alternatives involves so many unknowns, DOE considers it appropriate to provide a conservative estimate of the time that could be required both to develop the capability and to implement the alternative.

4. DOE did not include the estimated duration of operational readiness reviews or other preparatory actions for implementation of the alternatives. DOE expects such actions to be a prerequisite, regardless of the alternative selected (with the exception of No Action). However, DOE expects the level of environmental impacts from such activities to be the same as those estimated for the No-Action Alternatives because those activities primarily involve personnel training, maintenance, equipment checkout, and simulated operations. These are typical of activities that are ongoing at the SRS and are consistent with No Action within the context of the National Environmental Policy Act.

Specifically, DOE added information to Section 2.3.1 discussing the technical complexity and engineering challenges that it would have to overcome to transfer solutions containing significant amounts of fissile material (such as the plutonium-239 solutions stored in H-Canyon) to the SRS high-level waste tanks for vitrification in DWPF. This discussion provides the basis for DOE's estimates of 6 years for technical studies and a 3-year period during which DOE would transfer solutions. Sections 2.3.4, 2.3.5, 2.3.6, and 2.3.7 now describe a variation to the Improving Storage (Dry Storage) Alternative evaluated for fuel and targets stored in basins, including Mark-31 targets. DOE has examined ways of accelerating the design and construction schedule for such projects. The accelerated variation could result in a dry storage facility in approximately 5 years.

L11-10

Since 1945, every U.S. administration has recognized that preventing the further spread of nuclear weapons must be a fundamental national security and foreign policy objective of the United States. The current U.S. nuclear weapons nonproliferation policy is summarized in the *White House Fact Sheet on Nonproliferation and Export Control Policy*, dated September 27, 1993. Elements of this policy make clear that the United States does not encourage the civil use of plutonium and, accordingly, does not itself engage in plutonium reprocessing for either nuclear power or nuclear explosive purposes. In addition, United States policy will seek to eliminate where possible the accumulation of stockpiles of highly enriched uranium and plutonium.

With regard to the proposed actions in this EIS, any preferred alternative involving the recovery or chemical separation of highly enriched uranium also proposes to blend the material to low enriched uranium specifications, thereby eliminating the proliferation concern. Thus, such activities would be in concert with the nonproliferation policy of the United States.

None of the alternatives would create additional plutonium-239. Alternatives for the Mark-31 targets could result in the chemical separation of additional amounts of weapons-usable plutonium-239. DOE has committed to prohibit the use of this material for nuclear explosive purposes (DOE 1994c). Accordingly, such action would be consistent with U.S. nuclear nonproliferation policy.

L11-11

DOE agrees that cost considerations will play a role in the decisionmaking process and has prepared a separate report on the cost of implementing the various alternatives, 10-year integrated program costs for the four scenarios described in Chapter 5, and the waste costs associated with each scenario. Section 2.6 summarizes this report, which provides details on both capital (i.e., construction) and operating costs to implement possible combinations of the alternatives for each type of material. The cost reported has been added as a reference.

DOE agrees with the commentor that it is important to examine ways to expedite or compress the schedules and durations of stabilization activities and that doing so would be consistent with Departmental objectives in pollution prevention and waste minimization. DOE added a variation to the Improving Storage Alternative for fuel and targets that involves an accelerated design and construction schedule (see Sections 2.3.4, 2.3.5, 2.3.6, and 2.3.7). The use of conservative schedules in the EIS is not intended to imply that DOE would not make every effort to complete the required actions as quickly and efficiently as possible. However, DOE must also ensure that any required operations would be conducted in a very deliberate, controlled, and safe manner. It is DOE's intent to minimize the duration of stabilization activities through rigorous training and readiness reviews. DOE is undertaking a further management review of facility utilization plans, particularly as they apply to costs, U.S. nonproliferation policy, international inspections, and decisions to be made at other DOE sites. The findings from this study will be considered in the issuance of the Record of Decision(s) for SRS materials management.

For the purposes of estimating impacts, DOE used durations that reflect estimates based on past operating histories or engineering estimates and current requirements for conducting operations involving nuclear materials. The EIS does not attempt to establish a definitive schedule (i.e., exact start and finish dates) to accomplish each material stabilization activity. These schedules would depend on many factors, including availability of funding. However, DOE does have a proposed schedule for material stabilization based on the preferred alternatives identified in the EIS. The schedule is included in the Implementation Plan (IP) submitted by the Department to the Defense Nuclear Facility Safety Board (DNFSB) in response to DNFSB Recommendation 94-1 to the Secretary of Energy. The IP is publicly available in DOE reading rooms and is referenced in Chapter 1 of the EIS. DOE will further evaluate the cost implications of an integrated program for management of the materials (see Section 2.9).

L11-12

Section 2.2 of the Draft EIS indicated that the need to stabilize the programmatic materials was independent of the programmatic need for the material; that is, DOE would consider such materials candidates for stabilization even if there were no programmatic need. Therefore, the programmatic need for a material did not affect the decision to stabilize it but did restrict the reasonable alternatives for that material, because DOE preferred to implement an alternative in which the programmatic material would be recoverable for future use.

Since the issuance of the Draft EIS, DOE has decided that it is more appropriate to address long-term programmatic issues regarding plutonium-242 in the *Stockpile Stewardship and Management*

Programmatic Environmental Impact Statement, as discussed in the response to Comment L11-13. Further, DOE has determined that even though there could be a programmatic need for plutonium-242, the primary reason to stabilize this material is based on health and safety issues. Therefore, DOE has deleted the original (classified) Appendix B on the programmatic need for and use of plutonium-242 (and replaced it with a discussion of radioactive material and its hazards).

DOE received several public comments on both the americium and curium solutions and target material (see Comments L9, L10, L13, and L14). Based on these comments, DOE has included the target material as part of the americium and curium material in the programmatic category in this Final EIS, and has selected No Action as the preferred alternative for that material. DOE analyzed additional alternatives for all three programmatic materials, even though some of these alternatives (e.g., Processing and Storage for Vitrification in Defense Waste Processing Facility) would place the material in an unrecoverable form.

DOE interpreted the commentor's use of the term "unit cost" to mean the total costs associated with producing a given material on a per unit basis (e.g., $/g). DOE does not agree that the unit cost for obtaining these materials is relevant to the EIS. The material has already been produced by the irradiation of special targets in SRS reactors. With the exception of the obsolete targets, the material has already been dissolved and the programmatic isotopes chemically separated. The isotopes are currently stored in the chemical separations facilities in a purified liquid form. The costs related to the actions proposed by DOE in the EIS are those required to convert and store the materials in another physical form and to manage the resulting waste. As indicated in the response to Comment L11-11, DOE has summarized these costs in Section 2.6. The referenced cost report includes additional information on the processing, conversion, and storage of the programmatic materials. It is publicly available in DOE reading rooms along with the Final EIS.

L11-13

DOE has deleted Appendix B, "Programmatic Need for and Use of Plutonium-242," which appeared in the Draft EIS. DOE has decided to deal with long-term programmatic issues in an upcoming National Environmental Policy Act document, the *Stockpile Stewardship and Management Programmatic Environmental Impact Statement*, which is described in Section 1.6 of this Final EIS. Section 2.2 of this EIS contains descriptions of the reasonable stabilization alternatives DOE considered pending long-term decisions on programmatic need. In relation to the commentor's suggestion about making some information available to the public, DOE is performing a classification review, which is not yet complete. However, the results of this review should be available to support the decisions that DOE will make in relation to the Stockpile Stewardship Programmatic EIS.

L11-14

DOE has added information to Chapter 1 and Appendixes A and B on the hazards posed by the materials considered "candidates for stabilization." DOE has attempted to clarify the bases for concerns associated with each material. Chapter 1 references several in-depth reviews performed by the Department during the last 2 years. DOE relied extensively on these reviews in identifying "candidate materials for stabilization." The discussions in Chapter 1 on the concerns about the materials contain a significant number of direct quotations or excerpts from the ES&H reports. DOE considers the information contained in the EIS to be an accurate representation of the findings of the many technical experts who reviewed the conditions and hazards posed by these materials at the SRS. Consistent with other NEPA documents, the reports are included as part of the EIS by reference. As emphasized in the referenced reports, the experts considered continued exposure of the workers and the public to the current level of risk as a vulnerability or concern and recommended that DOE take actions to minimize or alleviate the hazard. The concern with a particular type of material could stem from:

1. The instability of a material due to its chemical composition

2. The inherent difficulties or undesirability of continuing to store a radioactive material in a particular physical form (e.g., large volumes of highly radioactive liquids)

3. The manner in which it is packaged or contained (e.g., plastic in contact with plutonium oxide or metal)

4. The environment to which the material or its storage container is being exposed (e.g., irradiated metal targets stored in single-walled water-filled basins)

5. Weaknesses or deficiencies associated with the facility in which the material is stored (e.g., inadequate ventilation system, inadequate fire protection system, nonexistence of monitoring or detection systems in the event of a leak, etc.)

DOE analyzed the operations of facilities in which the nuclear materials are currently stored. The risks associated with accidents involving continued storage of these materials have been quantified by DOE and are included in Appendix E of the EIS. Similarly, DOE estimated the risks associated with actions that it could take to stabilize these materials. This information is also included in Appendix E. There is uncertainty associated with any of these estimates. DOE believes that the estimates of consequences (i.e., dose and resulting potential health effects) are based on conservative assumptions. In other words, DOE believes the consequences tend to be overstated rather than underestimated. DOE acknowledges that the storage configurations for many of these materials are not consistent with the design intent (i.e., they have exceeded the intended duration). Therefore, the likelihood (i.e., frequency) of certain accidents could be increasing with time. In many cases, DOE has little operational experience on which to base any estimates of increased frequency for potential accidents. For example, DOE does not have a technical baseline on the extended storage of certain types of plutonium-bearing materials (i.e., radioactive solutions) and how chemical changes over time could contribute to an accident, such as a nuclear criticality. In essence, continued storage of many materials constitutes an ongoing experiment. DOE believes the lack of quantitative data emphasizes the need to rely instead on the collective judgment of the technical community. DOE believes that the conclusions presented in the EIS accurately reflect the consensus of the technical community that has reviewed continued storage of the materials. Where applicable, DOE has expanded qualitative discussions on the current risks posed.

L11-15

The commentor suggests that the inclusion of the irradiated reactor fuel and targets stored in H-Canyon as "candidates for stabilization" is illogical because the EIS states no corrosion has been observed. DOE did not base categorization of materials as candidates for stabilization solely on the basis of their current physical condition. As stated in the EIS, DOE considered the potential or likelihood that a material might pose a concern during the next 10 years. The fuel and targets stored in H-Canyon are of the same composition as those stored in the reactor disassembly basins (K-, L-, and P) that already have evidence of corrosion. In addition, the control of water chemistry in the H-Canyon basin is more difficult than at the reactor basins because the water is stagnant with no circulation or deionization capability. Water chemistry can have a direct effect in causing the fuel or targets to corrode in the H-Canyon basin. Although DOE expects to monitor the chemistry of the basin water in H-Canyon and to make periodic adjustments, the fuel and targets could experience some corrosion during the next 10 years. For these reasons, DOE considers it appropriate to categorize the materials as "candidates" for stabilization.

The commentor interpreted the DOE categorization of materials as "candidates for stabilization" as implying impending danger or hazards unless the Department takes immediate actions. Because DOE is evaluating the potential need for actions to be taken during an interim period (i.e., the next 10 years), DOE believes it is obligated to identify any storage situation that could reasonably pose a vulnerability concern.

L11-16

The commentor suggests that DOE has used inconsistent logic by stating the plutonium-242 solutions stored in H-Canyon and the americium and curium solutions stored in F-Canyon pose concerns even though they do not represent a nuclear criticality hazard (i.e., they are not fissile materials). These materials pose a concern due to their current physical form (i.e., liquid), the manner in which they are currently stored (single-walled stainless-steel tanks vented to the environment in the canyons), the relatively high concentration of radioactivity each liquid represents, and the potential for accidents resulting in a release of the material. Even though americium-243, curium-244, and plutonium-242 are not fissile isotopes, they are very high in specific activity when compared to most other radioisotopes stored at the SRS. Therefore, they represent a greater biological hazard than most other isotopes in the event of an accident. The characterization of these materials as a concern was based on all of these considerations, not just the potential for a nuclear criticality (or lack thereof). An expanded discussion on the hazards associated with the various forms of nuclear materials and the isotopes contained has been added to Chapter 1 and Appendixes A and B.

L11-17

DOE has expanded the discussion on Aging Facilities in Section 2.5.6 to include safety implications associated with the canyons and other facilities that might be required in the implementation of the alternatives.

Regarding the canyon ventilation system safety concern, DOE conducted safety reviews and readiness assessments to verify the adequacy of mitigative and emergency measures to respond to a ventilation system failure due to the potential collapse of the stack liner. In addition, prior to the restart or startup of the canyon facilities, readiness reviews have and will be performed (in accordance with DOE Order 5480.31, Startup and Restart of Nuclear Facilities) to ensure safety, regulatory, and process requirements are being achieved.

L11-18

This comment confuses two different cancer effects. The EIS addresses lifetime excess latent cancer fatalities, or the number of deaths expected sometime during the lifetime of a group of people due to exposure to ionizing radiation. The values quoted in the EIS are the excess cancer deaths that could be experienced from exposure to ionizing radiation and do not include cancer deaths from other causes. DOE selected this use of "excess cancer deaths" to make clear the expected effects from the materials under consideration and to avoid "diluting" these excess cancer deaths with the much larger number of cancer deaths expected from all causes. The EIS uses lifetime excess latent cancer fatalities to enable clear comparisons of one option to other options. DOE has added a discussion of radiation effects other than latent cancer effects to Chapter 4.

The use of a Dose Rate Effectiveness Factor (DREF) of 2 and the use of lifetime excess latent cancer fatality risk factors of 0.0005 fatality per person-rem for members of the public and 0.0004 fatality per person-rem for radiation workers are well documented in the scientific literature:

1. National Research Council, *Health Effects of Exposure to Low Levels of Ionizing Radiation, BEIR V*, 1990.

    a. Page 23, first paragraph: The DREF for leukemia of 2 is implicitly incorporated in leukemia risk values because of use of the linear-quadratic model for leukemia effects.

    b. Page 23, Table 1-4 shows that the DREF for solid cancers is in a range of 2 to 10, but the BEIR V Committee does not select a value. (Use of the factor of 2 is the *most conservative* value from the range they discuss.)

    c. Page 172, Table 4-2, footnote c notes the incorporation of an implicit DREF for leukemia risk, and footnote e documents that no DREF is incorporated in the solid cancer risks. The discussion of this table on page 174 references the range of DREFs (2 to 10) discussed in Chapter 1.

    d. Page 220, first paragraph discusses a DREF of 2. (A discussion of a factor of 2 increase for X-rays is not applicable to the exposures to IMNM materials because X-rays are not involved.)

    e. Page 237, Table 4F-5 shows a value of 1.92 to 1.99 for leukemia. The BEIR V Committee has rounded these values to the value of 2 used for leukemia.

2. International Commission on Radiological Protection, Publication 60, *1990 Recommendations of the International Commission on Radiological Protection*, November 1990.

    a. Page 18, paragraph (74) recommends a Dose and Dose Rate Effectiveness Factor (DDREF) of 2.

    b. Page 20, paragraph (83) recommends the use of a DDREF of 2, resulting in lifetime fatality probability coefficients of 0.04 and 0.05 per sievert for workers and the whole population, respectively. (These values are equivalent to 0.0004 and 0.0005 per person-rem).

c. Page 69, paragraph (S-9) specifies the use of a DDREF of 2.

d. Page 70, Table S-3 shows an excess latent cancer fatality risk factor per person-sievert of 0.04 for adult workers and 0.05 for the whole population. (These values are equivalent to 0.0004 and 0.0005 per person-rem).

e. Page 111, Annex B, Paragraph (B-62) discusses the DDREF and choices by other technical groups [UNSCEAR (1977) = 2 and 2.5, UNSCEAR (1986) = up to 5, UNSCEAR (1988b) = 2 to 10, NUREG (1989) = 3.3, US NIH (1985) = 2.3]. The ICRP recommends a value of 2.

3. National Council on Radiation Protection and Measurements, Report No. 115, *Risk Estimates for Radiation Protection*, December 1993.

   a. Page 104, Table 13.8 lists a total lifetime mortality in a population of all ages as 500×10^{-4} per sievert, and for a working population (footnote d) as 400×10^{-4} per sievert. (These values are equivalent to 0.0005 and 0.0004 per person-rem.)

   b. Page 112, Conclusions summarizes in bold-face type that the NCRP recommends using a DREF of 2 and that the risk factors of 0.04 per sievert and 0.05 per sievert be used for the worker population and the general population respectively. (These factors are equivalent to 0.004 and 0.005 per person-rem.)

4. National Radiological Protection Board, *Estimates of Late Radiation Risks to the U.K. Population*, Documents of the NRPB, Volume 4, No. 4, 1993, Oxon, England.

   a. The National Radiological Protection Board of the United Kingdom applies a DREF of 2.

5. Nuclear Regulatory Commission, NUREG/CR-4214, *Health Effects Models for Nuclear Power Plant Accident Consequence Analysis. Modifications of Models Resulting from Recent Reports on Health Effects of Ionizing Radiation. Low LET Radiation, Part II: Scientific Basis for Health Effects Models*, Revision 1, Part II, Addendum 1, LMF-132, 1991.

   a. The Nuclear Regulatory Commission uses a DREF of 2.

6. Nuclear Regulatory Commission, *Code of Federal Regulations, Part 20, Federal Register*, Vol. 56, No. 98, pages 23360-23474, May 21, 1991.

    a. Page 23363 of the *Federal Register*, Section H specifies the lifetime risk of radiation induced cancer fatalities is about 0.0004 fatal cancer per rem for workers and 0.0005 fatal cancer per rem for the general population.

7. Committee on Interagency Radiation Research and Policy Coordination, Office of Science and Technology Policy, Executive Office of the President, *Use of BEIR V and UNSCEAR 1988 in Radiation Risk Assessment*, Science Panel Report No. 9, December 1992.

    a. Page 8: The Committee on Interagency Radiation Research and Policy Coordination (CIRRPC) recommends that Federal agencies use a DREF of 2.

    b. Page 9: CIRRPC recommends that Federal agencies use a nominal risk estimate of 4.5×10^{-3} for the lifetime total cancer mortality for the general population at the reference whole-body absorbed dose of 10 rad. (Equivalent to 0.00045 per person-rem.)

    c. Page 10: CIRRPC recommends that Federal agencies use a nominal risk estimate of 3.5×10^{-3} for the lifetime total cancer mortality for the working-age population at the reference whole-body absorbed dose of 10 rad. (Equivalent to 0.00035 per person-rem.)

L11-19

Chapter 4 of the Draft EIS states that the alternatives under any of the analyzed scenarios are well within the 10-millirem-per-year dose limit for offsite impacts as imposed by DOE Order 5400.5 and the Clean Air Act. Chapter 4 of the EIS has been revised to add tables directly comparing each of the alternatives relative to air emissions. As can be seen from Tables 4-1 through 4-11, none of the alternatives exceeds 3.3 millirem. Similarly, Chapter 5 presents cumulative impacts from air emissions. As can be seen from Chapter 5, none of DOE's contemplated actions under any scenario is likely to exceed 40 percent of the 10 millirem limit. This has been clarified in the respective sections of Chapters 4 and 5.

Chapter 4 also discusses compliance with the nonradiological requirements of the Clean Air Act. Compliance with the ambient air quality standards (both Federal and State) and toxic air pollutant regulations is discussed in Section 3.3.3. Tables 3-5 and 3-6 indicate that existing SRS operations meet the criteria for ambient air standards and toxic air pollutants. Chapter 4 has been revised to include tables that directly compare each alternative to the applicable standards. The comparisons presented in Chapters 4 and 5 indicate that no air standard is likely to be exceeded under any scenario.

L11-20

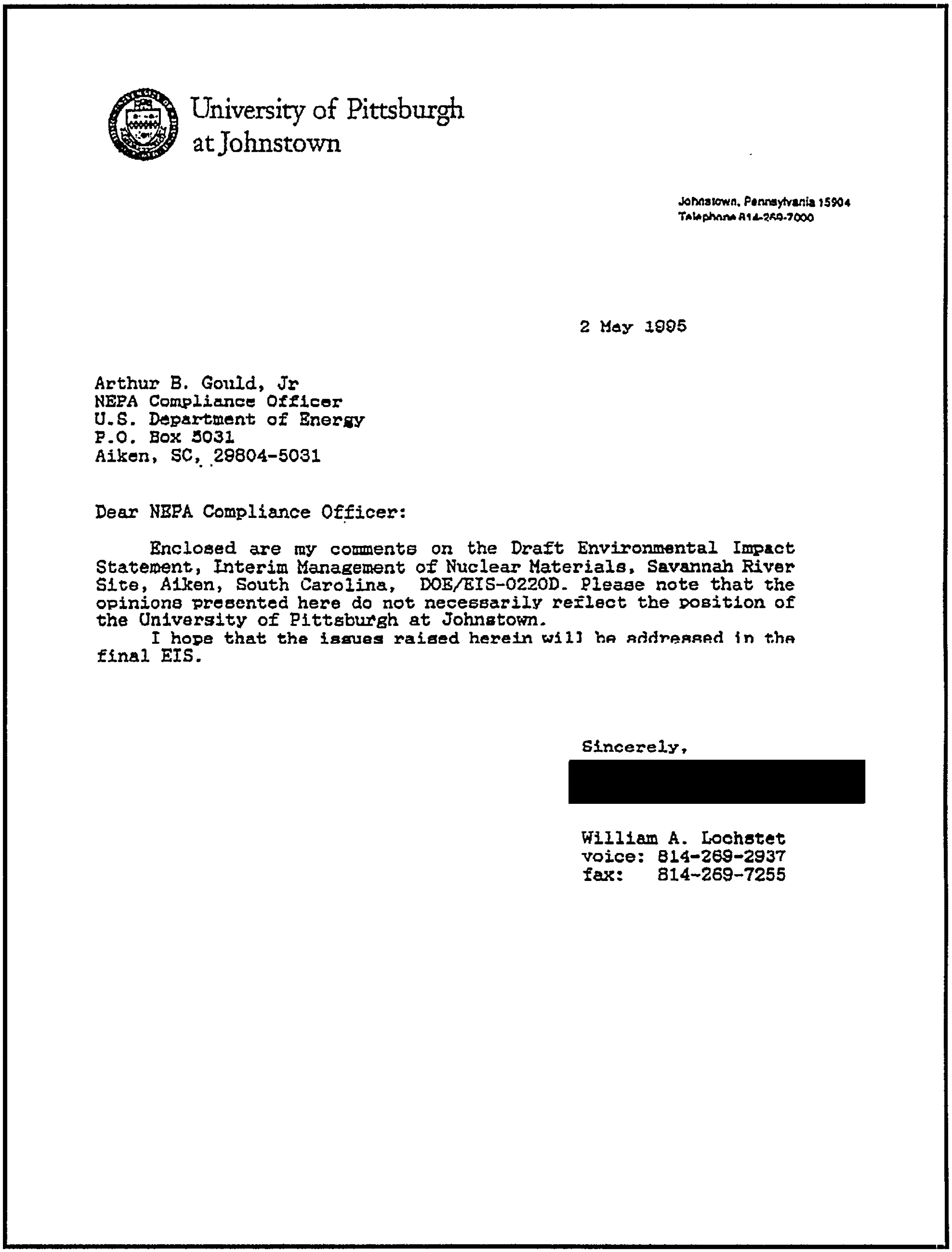
In a letter dated January 13, 1995, DOE asked the South Carolina Department of Health and Environmental Control, the State agency with authority to administer the Resource Conservation and Recovery Act, for concurrence that plutonium solutions in F-Canyon are "in process" and therefore not waste. Specifically, the question was whether the nitric acid that is associated with the solutions should be considered waste. The Department stated that the nitric acid continues to serve the same criticality/radiological safety function of maintaining the plutonium in a solution pending stabilization. On January 26, 1995, DOE received a letter of concurrence from SCDHEC on the regulatory status of the F-Canyon plutonium solutions. Although the letter is specific to F-Canyon solutions, the Department maintains that the same regulatory interpretation applies to H-Canyon solutions and other solutions in F-Canyon. On July 7, 1995, DOE sent a letter to SCDHEC on the materials that are the subject of this EIS. In its response of August 8, 1995, SCDHEC agreed with the DOE position that the solutions evaluated in this EIS are not currently subject to regulation as mixed waste. DOE and SCDHEC are continuing to consult on the regulatory status of the remaining SRS materials. DOE recently submitted a Proposed Site Treatment Plan (PSTP) to SCDHEC as part of its compliance with the Federal Facility Compliance Act. The PSTP discusses proposed treatment of mixed wastes, as defined by RCRA. None of the nuclear materials included in the scope of the EIS was identified as waste in the PSTP.

L11-21

Consistent with the commentor's suggestion, DOE has included Tables 4-1 through 4-11 in Chapter 4 summarizing the 10-year impacts (based on annual data contained in Appendix D) to provide the reader with a direct comparison of the air emissions for each alternative. (This was done for selected air pollutants and radioactive emissions in Chapter 2 of the Draft EIS).

L11-22

Consistent with the commentor's suggestion, DOE has included Tables 4-1 through 4-11 in Chapter 4 and Tables 5-2 and 5-12 in Chapter 5. The tables in Chapter 4 provide a direct comparison among the alternatives for a given material type. Similarly, the tables in Chapter 5 compare the cumulative impacts expected from various combinations of alternatives (scenarios), as well as the waste impacts expected from all proposed DOE actions potentially involving the SRS.



University of Pittsburgh
at Johnstown

Johnstown, Pennsylvania 15904
Telephone 814-269-7000

2 May 1995

Arthur B. Gould, Jr
NEPA Compliance Officer
U.S. Department of Energy
P.O. Box 5031
Aiken, SC, 29804-5031

Dear NEPA Compliance Officer:

Enclosed are my comments on the Draft Environmental Impact Statement, Interim Management of Nuclear Materials, Savannah River Site, Aiken, South Carolina, DOE/EIS-0220D. Please note that the opinions presented here do not necessarily reflect the position of the University of Pittsburgh at Johnstown.

I hope that the issues raised herein will be addressed in the final EIS.

Sincerely,

[redacted]

William A. Lochstet
voice: 814-269-2937
fax: 814-269-7255

Comment L12. Page 1 of 3.

Some Suggestions on Interim Management of Nuclear Materials

by

Wm. A. Lochstet
The University of Pittsburgh
at Johnstown*
May 1995

The Department of Energy (DOE) has prepared a Draft Environmental Impact Statement on the Interim Management of Nuclear Materials, DOE/EIS-0220D. This document evaluates the environmental impacts of alternatives for the next 10 years, for storage of nuclear materials at the Savannah River Site (SRS).

L12-1 The Draft states that there are about 14,000 liters of americium and curium solution stored in a single tank in the F-Canyon. The continued storage in liquid form is a tank waiting for a leak, and is not acceptably stable. The DOE (section 2.2.2) properly concludes that this material should be converted to a solid form. Two alternatives considered are vitrification and conversion to oxide. Vitrification would produce a glass material which would be stored in canisters. This product would be difficult to dissolve or use at any point in the future. Considering the possible uses of this material, (as outlined in section 2.2.2), it would seem more appropriate to store it as solid oxide. One problem cited for the oxide alternative is that loading and packaging would be awkward due to the high radiation level of 30 rem per hour at 1 m. However, the radiation level for vitrification is described as three times as great. Clearly this can be varied by changing the canister size. Since this is an interim decision, it would seem that options for the future should be preserved, by converting this material to oxide form.

L12-2 The Draft describes its radiological accident analysis method in section E.2.2 of Appendix E. The last paragraph of this section describes a "typical calculation" as an example of an accident in which a worker drops a container holding 2 kg of plutonium oxide powder. One must assume that some fraction of this material becomes airborne and respirable. To make a first calculation, one assumes that the material instantly disperses throughout the air in the room. It is also necessary to choose a particular room size, and the time before the worker escapes. These factors will determine the amount of oxide deposited in the lungs. DOE finds that the calculated dose would be between 80 and 78,000 rem. The description concludes that this is a minor event with only minor personnel contamination and assimilation. This is a gross understatement. It is suggested that something is VERY WRONG if this is considered MINOR. It is also contradictory to suggest that a dose of 78,000 rem is minor.

* Affiliation for identification purposes only.

Appendix A is a list of nuclear materials at SRS. In this list at page A-7 is a description of "Scrap and residue plutonium solids" which includes the statement "oxides of plutonium-239 having other actinide impurities other than americium-241, the decay daughter of plutonium-239." This is a clear misprint. plutonium-239 does dot decay to americium-241. L12-3

-end-

Comment L12. Page 3 of 3.

Response to comment letter L12:

L12-1

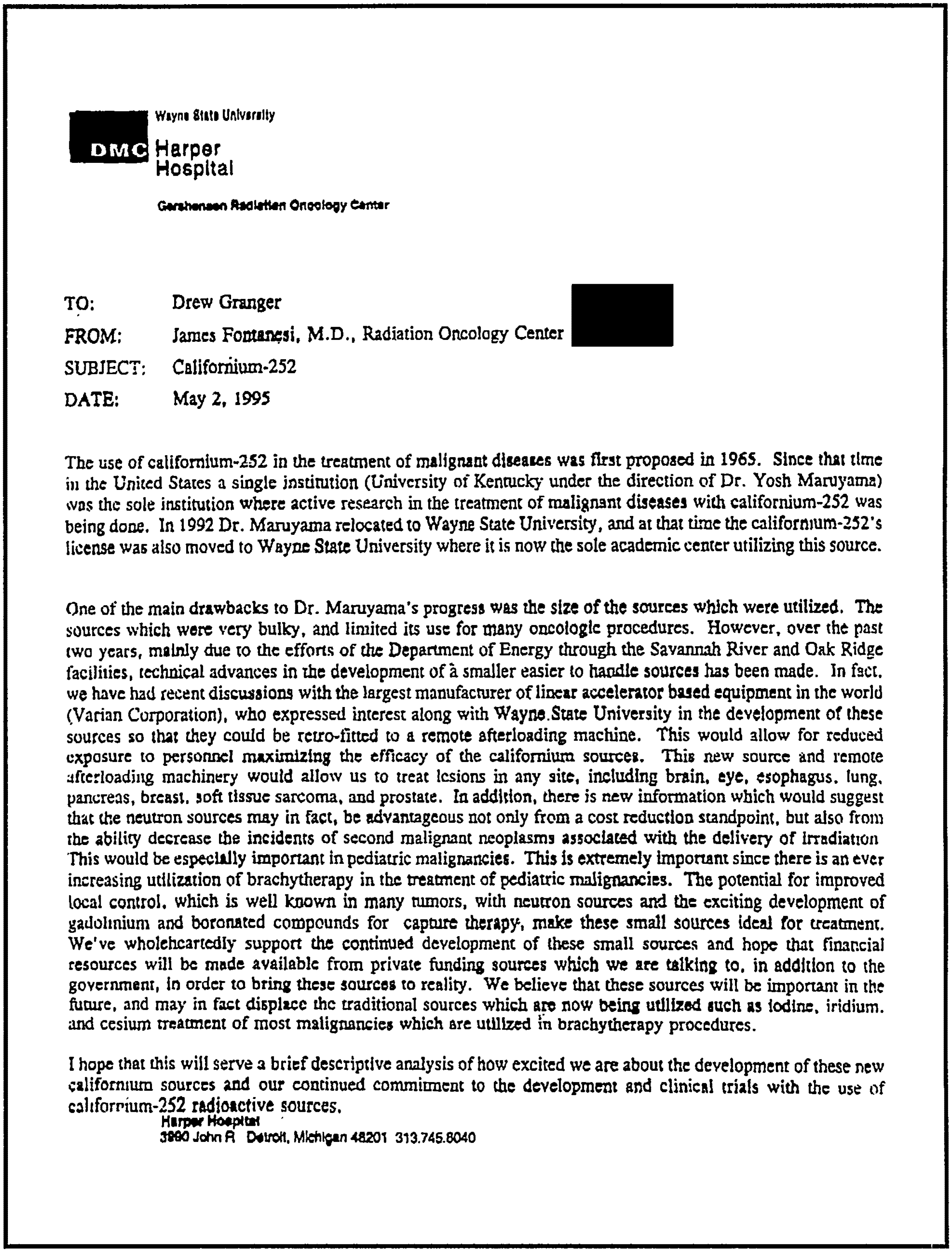
Vitrification would not hinder or eliminate future options or uses for the americium and curium. In fact, potential users of these materials have indicated a preference for this solid glass form. Further, vitrification would be likely to reduce impacts during handling and shipping. Vitrification does not have to be a final phase for nuclear material for which there are alternative uses.

L12-2

The text in Section E.2.2 has been modified to clarify the intent of describing the potential range of consequences. DOE did not wish to imply that personnel doses, even at the lower end of the calculated range, would be minor.

L12-3

The commentor is correct. Appendix A has been modified to reflect this comment.



Wayne State University
DMC Harper
Hospital

Gershenson Radiation Oncology Center

TO: Drew Granger
FROM: James Fontanesi, M.D., Radiation Oncology Center [redacted]
SUBJECT: Californium-252
DATE: May 2, 1995

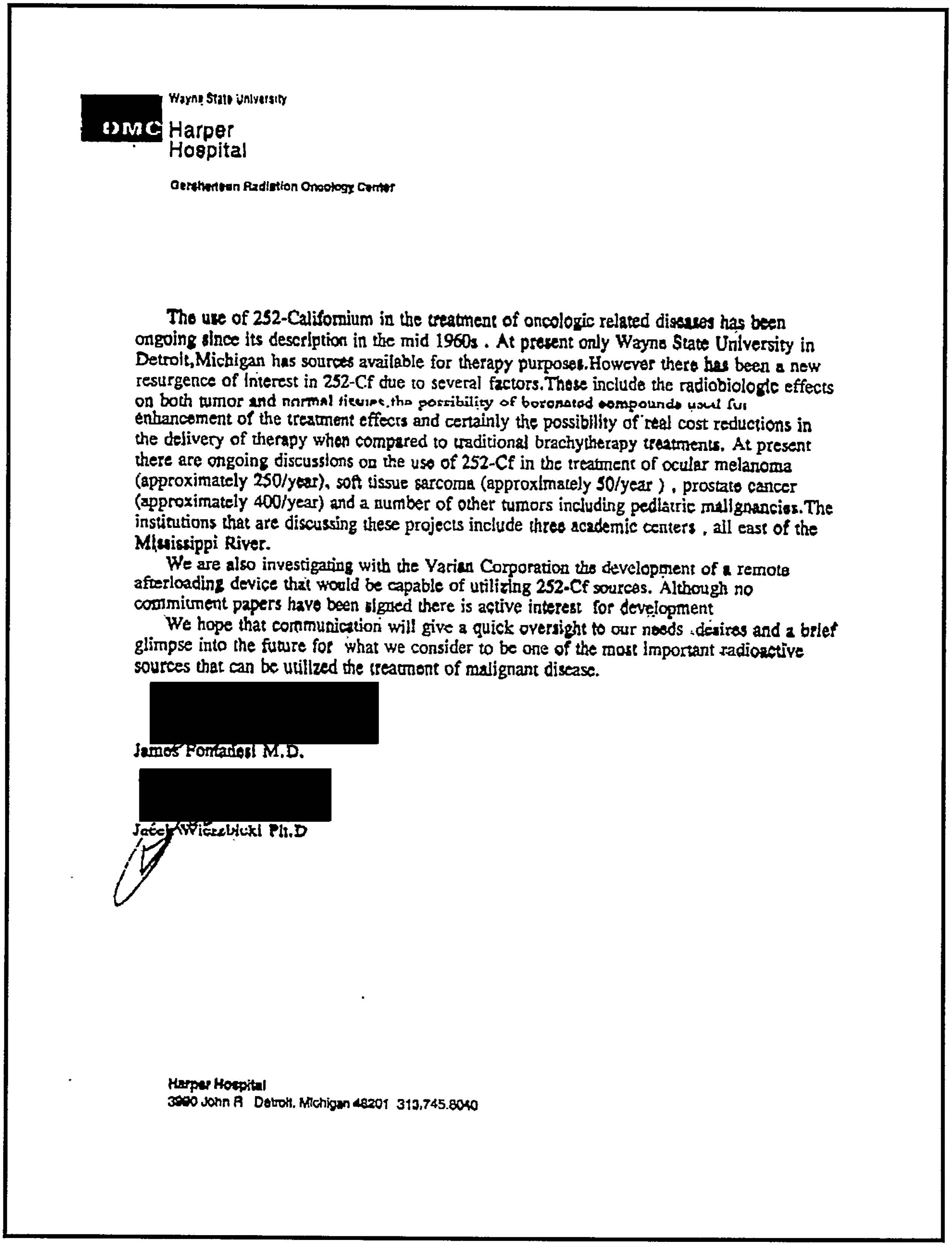
The use of californium-252 in the treatment of malignant diseases was first proposed in 1965. Since that time in the United States a single institution (University of Kentucky under the direction of Dr. Yosh Maruyama) was the sole institution where active research in the treatment of malignant diseases with californium-252 was being done. In 1992 Dr. Maruyama relocated to Wayne State University, and at that time the californium-252's license was also moved to Wayne State University where it is now the sole academic center utilizing this source.

One of the main drawbacks to Dr. Maruyama's progress was the size of the sources which were utilized. The sources which were very bulky, and limited its use for many oncologic procedures. However, over the past two years, mainly due to the efforts of the Department of Energy through the Savannah River and Oak Ridge facilities, technical advances in the development of a smaller easier to handle sources has been made. In fact, we have had recent discussions with the largest manufacturer of linear accelerator based equipment in the world (Varian Corporation), who expressed interest along with Wayne State University in the development of these sources so that they could be retro-fitted to a remote afterloading machine. This would allow for reduced exposure to personnel maximizing the efficacy of the californium sources. This new source and remote afterloading machinery would allow us to treat lesions in any site, including brain, eye, esophagus, lung, pancreas, breast, soft tissue sarcoma, and prostate. In addition, there is new information which would suggest that the neutron sources may in fact, be advantageous not only from a cost reduction standpoint, but also from the ability decrease the incidents of second malignant neoplasms associated with the delivery of irradiation This would be especially important in pediatric malignancies. This is extremely important since there is an ever increasing utilization of brachytherapy in the treatment of pediatric malignancies. The potential for improved local control, which is well known in many tumors, with neutron sources and the exciting development of gadolinium and boronated compounds for capture therapy, make these small sources ideal for treatment. We've wholeheartedly support the continued development of these small sources and hope that financial resources will be made available from private funding sources which we are talking to, in addition to the government, in order to bring these sources to reality. We believe that these sources will be important in the future, and may in fact displace the traditional sources which are now being utilized such as iodine, iridium, and cesium treatment of most malignancies which are utilized in brachytherapy procedures. L13-1

I hope that this will serve a brief descriptive analysis of how excited we are about the development of these new californium sources and our continued commitment to the development and clinical trials with the use of californium-252 radioactive sources.

Harper Hospital
3990 John R Detroit, Michigan 48201 313.745.8040

Comment L13. Page 1 of 2.



Wayne State University
DMC Harper
Hospital

Gershenson Radiation Oncology Center

L13-1 The use of 252-Californium in the treatment of oncologic related diseases has been ongoing since its description in the mid 1960s . At present only Wayne State University in Detroit,Michigan has sources available for therapy purposes.However there has been a new resurgence of interest in 252-Cf due to several factors.These include the radiobiologic effects on both tumor and normal tissues,the possibility of boronated compounds used for enhancement of the treatment effects and certainly the possibility of real cost reductions in the delivery of therapy when compared to traditional brachytherapy treatments. At present there are ongoing discussions on the use of 252-Cf in the treatment of ocular melanoma (approximately 250/year), soft tissue sarcoma (approximately 50/year) , prostate cancer (approximately 400/year) and a number of other tumors including pediatric malignancies.The institutions that are discussing these projects include three academic centers , all east of the Mississippi River.

We are also investigating with the Varian Corporation the development of a remote afterloading device that would be capable of utilizing 252-Cf sources. Although no commimment papers have been signed there is active interest for development

We hope that communication will give a quick oversight to our needs ,desires and a brief glimpse into the future for what we consider to be one of the most important radioactive sources that can be utilized the treatment of malignant disease.

[redacted]

James Fontanesi M.D.

[redacted]

Jacek Wierzbicki Ph.D

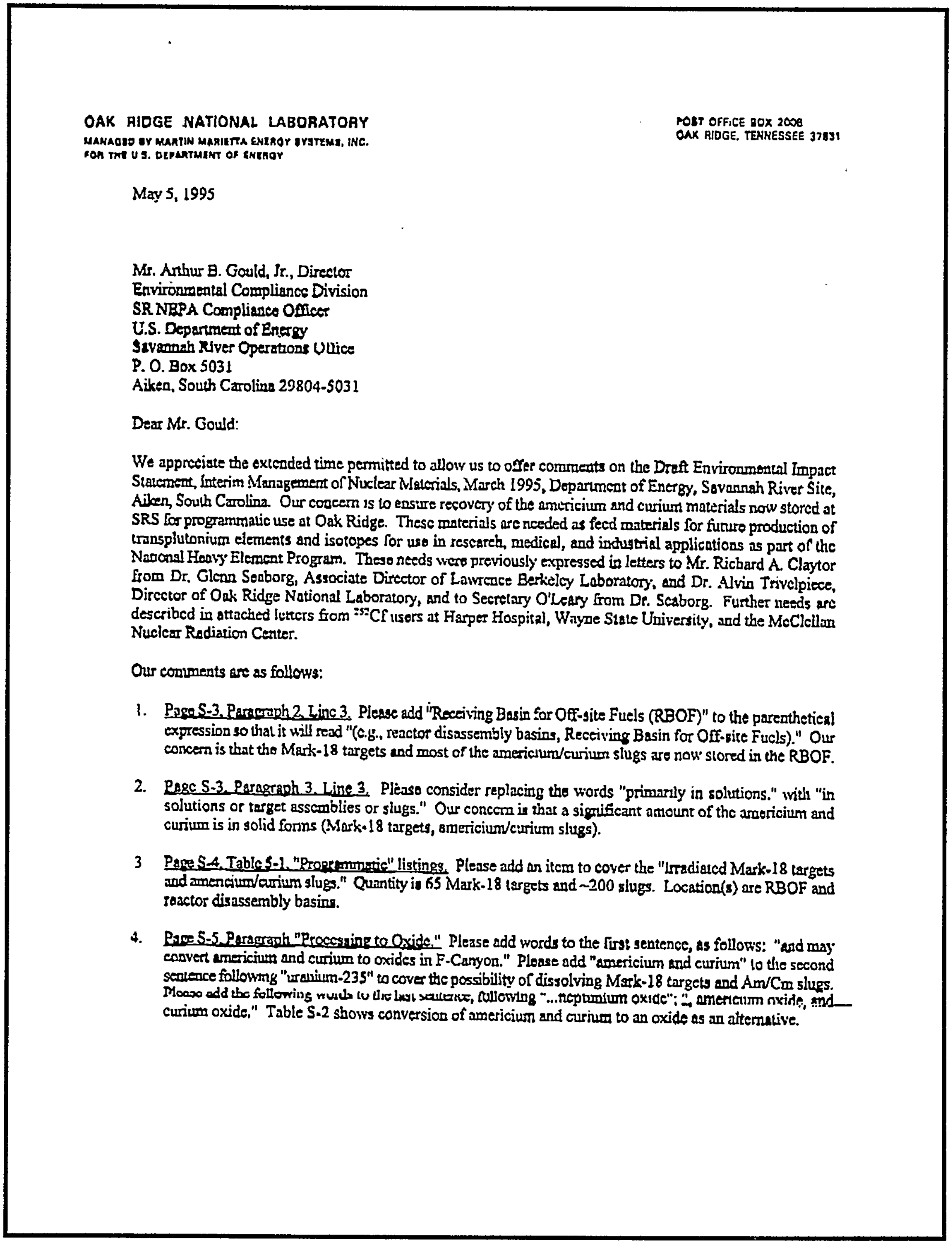
Harper Hospital
3990 John R Detroit, Michigan 48201 313.745.8040

Comment L13. Page 2 of 2.

Response to comment letter L13:

L13-1

DOE understands the need for californium in medical applications and medical research; the revised Section 2.2.2 of the EIS recognizes these uses.



OAK RIDGE NATIONAL LABORATORY
MANAGED BY MARTIN MARIETTA ENERGY SYSTEMS, INC.
FOR THE U.S. DEPARTMENT OF ENERGY

POST OFFICE BOX 2008
OAK RIDGE, TENNESSEE 37831

May 5, 1995

Mr. Arthur B. Gould, Jr., Director
Environmental Compliance Division
SR NEPA Compliance Officer
U.S. Department of Energy
Savannah River Operations Office
P. O. Box 5031
Aiken, South Carolina 29804-5031

Dear Mr. Gould:

L14-1 We appreciate the extended time permitted to allow us to offer comments on the Draft Environmental Impact Statement, Interim Management of Nuclear Materials, March 1995, Department of Energy, Savannah River Site, Aiken, South Carolina. Our concern is to ensure recovery of the americium and curium materials now stored at SRS for programmatic use at Oak Ridge. These materials are needed as feed materials for future production of transplutonium elements and isotopes for use in research, medical, and industrial applications as part of the National Heavy Element Program. These needs were previously expressed in letters to Mr. Richard A. Claytor from Dr. Glenn Seaborg, Associate Director of Lawrence Berkeley Laboratory, and Dr. Alvin Trivelpiece, Director of Oak Ridge National Laboratory, and to Secretary O'Leary from Dr. Seaborg. Further needs are described in attached letters from ^{252}Cf users at Harper Hospital, Wayne State University, and the McClellan Nuclear Radiation Center.

Our comments are as follows:

L14-2
1. Page S-3, Paragraph 2, Line 3. Please add "Receiving Basin for Off-site Fuels (RBOF)" to the parenthetical expression so that it will read "(e.g., reactor disassembly basins, Receiving Basin for Off-site Fuels)." Our concern is that the Mark-18 targets and most of the americium/curium slugs are now stored in the RBOF.

L14-3
2. Page S-3, Paragraph 3, Line 3. Please consider replacing the words "primarily in solutions." with "in solutions or target assemblies or slugs." Our concern is that a significant amount of the americium and curium is in solid forms (Mark-18 targets, americium/curium slugs).

L14-4
3. Page S-4, Table S-1, "Programmatic" listings. Please add an item to cover the "Irradiated Mark-18 targets and americium/curium slugs." Quantity is 65 Mark-18 targets and ~200 slugs. Location(s) are RBOF and reactor disassembly basins.

L14-5
4. Page S-5, Paragraph "Processing to Oxide." Please add words to the first sentence, as follows: "and may convert americium and curium to oxides in F-Canyon." Please add "americium and curium" to the second sentence following "uranium-235" to cover the possibility of dissolving Mark-18 targets and Am/Cm slugs. Please add the following words to the last sentence, following "...neptunium oxide": ", americium oxide, and curium oxide." Table S-2 shows conversion of americium and curium to an oxide as an alternative.

Comment L14. Page 1 of 2.

5. Page S-7, Paragraph "Vitrification in F-Canyon." Please change the third sentence to read "The canisters would be placed in storage in the canyon, RBOF, heavily-shielded casks or vaults." L14-6

6. Page 1-16, 4th Paragraph, Line 7. Please add the following words to the sentence, following "...reactor disassembly basins": "and the receiving basin for off-site fuels." Please consider the deletion of "unusable." It carries an undefined connotation. This paragraph is a good, short description of the programmatic need. L14-7

7. Page 1-17, Table 1-1. Same comments as item 3 above, to add Mark-18 targets and americium/curium slugs to the "Programmatic" category. L14-8

8. Page 2-7, Item 2.2.2 AMERICIUM AND CURIUM, Paragraph 1, Line 3. Please delete the words "National Heavy Metal and Advanced Neutron Source Program" and replace with "National Heavy Element Program."

   Please add a paragraph to include the Mark-18 targets and americium/curium slugs. The following is suggested: "Sixty-five Mark-18 targets and ~200 americium/curium slugs are stored in the Receiving Basin for Off-Site Fuels and the reactor disassembly basins. These materials contain additional amounts of valuable americium and curium. These materials will be processed or converted to a suitable solid form for transportation, storage, and eventual use at Oak Ridge." L14-9

9. Appendix A, Table A-1. Please delete the Mark-18 targets listed under "Stable Material" on page A-2. Please add an item under "Programmatic Materials" on page A-5 under the sub-heading "Americium and Curium," as follows. "Targets - Sixty-five irradiated Mark-18 targets and ~200 irradiated americium/curium targets are stored underwater in RBOF and the reactor disassembly basins. Routinely monitored and inventoried. RBOF, Reactor Disassembly Building 244 L14-10

We appreciate your consideration of our comments. If you have any questions about them, please call me, John Bigelow, or Chuck Alexander at (615) 574-6928.

Sincerely,

[redacted]

E. D. Collins, Head
Isotope Technology Section
Chemical Technology Division

EDC:jp

Attachments

c: C. W. Alexander
J. E. Bigelow
L. E. McNeese
REDC-DCC - RC
File

Comment L14. Page 2 of 2.

Response to comment letter L14:

L14-1

Comment noted. DOE always tries to accommodate comments received after the end of the public comment period.

L14-2

The comment refers to a paragraph that describes the reasons DOE considers certain materials to be candidates for stabilization. It would not be appropriate to add the Receiving Basin for Offsite Fuel (RBOF) as the commentor suggests because DOE believes the RBOF facility can provide safe storage for fuel and target materials for extended periods. However, the targets stored in RBOF have been added to the list of programmatic materials throughout the EIS. In addition, the Taiwan Research Reactor and Experimental Breeder Reactor-II fuel have been categorized as candidates for stabilization due to their poor condition, as discussed in the response to L3-1.

L14-3

The Summary has been modified in response to this comment.

L14-4

In response to this and similar comments, DOE has changed Table S-1, Table 1-1, and Appendix A to identify specifically the Mark-18 targets in the Receiving Basin for Offsite Fuel and 60 Mark-51 slugs in the P-Reactor disassembly basin.

L14-5

DOE has not modified this paragraph because it represents the preferred alternative for the management of the various nuclear materials. Table S-2 has been changed to list the preferred alternatives of vitrifying the liquid americium and curium solutions in F-Canyon and continuing storage (No Action) of the Mark-18 targets and Mark-51 slugs. The conversion of all americium and curium materials is listed as an alternative.

L14-6

DOE would not propose to use storage space in the Receiving Basin for Offsite Fuel to store canisters containing the vitrified americium and curium solutions, so this paragraph has not changed.

L14-7

DOE agrees that the use of the word "unusable" is confusing.

L14-8

See the response to Comment L14-4.

L14-9

DOE has modified this paragraph to include the correct program title and acknowledge the presence of the Mark-18 targets in the Receiving Basin for Offsite Fuel and the Mark-51 slugs in the P-Reactor disassembly basin.

L14-10

DOE has modified Table A-1 as suggested to include the Mark-18 targets in the Receiving Basin for Offsite Fuel, the 60 slugs in the P-Reactor disassembly basin, and the 114 slugs in RBOF as programmatic material. However, at this time DOE's preferred alternative for the target and slug material is Continuing Storage (No Action) rather than processing because, in contrast to the solutions in F-Canyon, DOE believes that the solid materials can be safely stored without processing.

INTERIM MANAGEMENT OF NUCLEAR MATERIALS
ENVIRONMENTAL IMPACT STATEMENT
STATEMENT FOR THE RECORD

Number one, my comment on the Interim Management of Nuclear Materials EIS Draft is, first of all, that I'm a scientist in Nondestructive Evaluation Department for Mason Hangar at the Pantex Plant near Amarillo, Texas. We do neutron radiography on small components of nuclear weapons. These components have, in many cases, explosives encased in metal, making them not suitable for normal X-ray. Within the next year, our new neutron radiography facility will require 150 milligrams of californium-252. Because of decay each year, we will need new source capsules containing about 53 milligrams of californium-252. The californium-252 will come from Oak Ridge National Laboratory using the HFIR reactor and the Radiochemical Engineering Center. They expect to use as feed materials, at least in part, the americium and curium now in F-Canyon at the Savannah River Site.

Thank you for including this input. Okay. This is all. Thank you.

Wayne Sievers
Nondestructive Evaluation
Mason & Hangar Co.
Amarillo, Texas 79177

Voice-mail comment V1.

Response to comment V1:

V1-1

DOE thanks the commentor for taking the time to comment on this EIS. In Section 2.2.2, DOE acknowledges the programmatic need for californium.

INTERIM MANAGEMENT OF NUCLEAR MATERIALS

ENVIRONMENTAL IMPACT STATEMENT

STATEMENT FOR THE RECORD

This comment was given at the April 11, 1995, meeting in Savannah, Georgia:

For the record I'd like to continue to reiterate my firm belief, as I've done in the past, that there's absolutely no way--I repeat, absolutely no way--that the Savannah River Site can operate safely. It is a hazard not only to Savannah but to the whole region. It's a falsehood to believe that anything can be buried at that site and not have it leak into the groundwater. The technology is not available and the Department of Energy is just pulling the wool over our eyes in having us believe that this is so.

As far as the need for the Savannah River Site, it's totally unnecessary. As has been acknowledged in this room that the Cold War is over. Everybody knows that. And the need to maintain a large nuclear arsenal or any nuclear arsenal for that matter is totally unnecessary. We all--we still have a large nuclear arsenal
H1-1 unfortunately. And there's absolutely no reason to maintain a reprocessing center in the Savannah River Site. It should be totally abolished and converted into something more useful like a national park. If it can ever be cleaned up, and I hope it can be cleaned up some day, the Savannah River Site will serve a purpose, but
H1-2 right now the Government has not spent enough to clean it up and it still wants to use the site to produce more nuclear materials and reprocess the existing ones. We don't need more tritium; we don't need more plutonium. We already have enough and we certainly don't need more highly enriched uranium.

Comment H1. Page 1 of 2.

The byproducts enter the Savannah River more than we'd like to realize. Sometimes it makes the newspaper. But cancer in this community is rampant and there's no reason to believe that the Savannah River--the existence of the Savannah River Site is not a strong contributor. I want to alarm everybody that what we are talking about here is a community hazard just to preserve a few jobs. I would like to see the Savannah River Site closed and remain closed. The Department of Energy is just trying to convince us that it's--it is perfectly safe, but it is not safe. This is a falsehood. All the propaganda that's being put out in this room now does not refute the fact that dangerous materials exist at that site. And these materials are poisoning the whole Savannah community and the whole South Carolina-Georgia region as well. It escapes into the air and into the ground water. Besides the fact that it is over an earthquake fault, the Savannah River Site continues to exist as though nothing will ever happen to it, that the community is totally safe, that it's just a sheep farm. Well, go ahead and believe that if you want to pull the wool over your eyes, but everything said in this room today should be seen as something out of the *Wizard of Oz*.

The only way -- if the only way to store it is in -- to store plutonium and other nuclear materials is in glass cases, in glass rock, what happens when glass cracks? Do you honestly believe that this glass will never crack buried deep in the ground? I don't. The earth is constantly moving. The earth is constantly changing. We don't need to take the risk of burying more and more nuclear materials underground at the Savannah River Site. I don't know what the answer is, but the answer is not to poison the Savannah community. Thank you. H1-3

Fred Nadelman
1825 E. Gwinnett Street
Savannah, GA 31404

Comment H1. Page 2 of 2.

Response to oral comment H1:

H1-1

DOE suspended processing operations in the F- (and H-) Canyon in 1992 to address a potential safety concern. The Record of Decision for the F-Canyon Plutonium Solutions EIS approved the resumption of some processing activities to stabilize plutonium-bearing solutions in tanks in F-Canyon.

H1-2

DOE will address the issues associated with waste management, environmental restoration, and cleanup activities in programmatic and site-specific waste management EISs.

H1-3

The underground storage of plutonium is a long-term disposition issue. DOE is addressing this issue in its EIS on Storage and Disposition of Weapons-Usable Fissile Nuclear Materials, which is in preparation.

INTERIM MANAGEMENT OF NUCLEAR MATERIALS

ENVIRONMENTAL IMPACT STATEMENT

STATEMENT FOR THE RECORD

This comment was given at the April 11, 1995, meeting in Savannah, Georgia:

- A study says the highest exposure of radiation in the F-Canyon area is due to a tritium plume underneath the Chemical Separations Facility. H2-1
- What will vitrification in the F-Canyon cost? H2-2
- If it is decided to build an F-Canyon vitrification facility, would the alternatives for stabilization of materials change? Is it now going to be the primary storage or disposal method? H3-3
- Do other sites have americium and curium? H2-4

George Minot
3 Bateau Road
Hilton Head Island, SC 29928-3012

Comment H2 (summarization).

Response to oral comment H2:

H2-1

DOE acknowledges that tritium plumes that result from the historic use of seepage basins could exist in areas near the canyons. However, the tritium in the groundwater is not the direct result of canyon operations, but is the subject of separate environmental restoration programs at the SRS.

H2-2

DOE has prepared a cost report that will be available at the same time as the Final Interim Management of Nuclear Materials EIS. This report will contain reasonable estimates of the capital costs associated with each alternative and the operating expenses that each alternative would incur throughout its lifetime.

H2-3

Any decisions on the stabilization of the materials discussed in this EIS will be independent decisions. DOE constructed this EIS to enable the decisionmaker to choose any of the described alternatives for a given material without impacting decisions about other materials. Therefore, the choice of Vitrification (F-Canyon) as the stabilization technique for americium and curium would not dictate the stabilization of other materials in the same manner. On the other hand, the decisionmaker could choose Vitrification (F-Canyon) to stabilize any of the materials for which that alternative would be reasonable.

H2-4

Yes, there are small quantities of americium and curium in inventory at DOE national laboratories, such as, Oak Ridge in Tennessee and Lawrence Livermore in California. The americium and curium materials stored at the SRS represent valuable feedstock for the activities conducted by these laboratories. The SRS materials contain a large percentage of the world supply of many rare and important heavy isotopes (Seaborg to Claytor letter, December 16, 1992).

INTERIM MANAGEMENT OF NUCLEAR MATERIALS

ENVIRONMENTAL IMPACT STATEMENT

STATEMENT FOR THE RECORD

This comment was given at the April 11, 1995, meeting in Savannah, Georgia:

For the record I would just like to say that a number of residents in Savannah feel that ten years is -- is too long for those actions to occur at the Savannah River Site and they're asking if there could be a possible reduction of that time. H3-1

Under the alternatives of proposed action -- and that's a -- this is another comment --people are feeling that they need more help in understanding the issue in order to be able to respond to the action and they are asking if that can be made available more basic workshops where they could begin to really understand. We're talking about doing these things for a long period of time so our young people will be growing up living in the area. And we feel that it's important for ourselves as well as our young people to get at least a basic understanding of the issue. So the community is asking if there could be possibly some type of basic educational sessions that can help us to understand the topic and the issues of our nuclear materials. H3-2

Qasimah P. Boston
Citizens for Environmental Justice
1115 Habersham Street
Savannah, GA 31401

Comment H3.

Response to oral comment H3:

H3-1

DOE decided to increase the length of time to make a decision and implement stabilization methods for nuclear materials from 5 years, as stated in the Notice of Intent to prepare this EIS, to 10 years in response to a significant number of suggestions from members of the public. The public's perception was that 5 years might -- in some cases and depending on the nuclear material -- be optimistic. DOE does not expect to take 10 years to make the decisions and implement any of the actions described in this EIS.

H3-2

DOE has an ongoing public outreach program at the Savannah River Operations Office, and considers feedback such as this important to the program's success. DOE will base any improvement to outreach efforts on lessons learned and continuous program evaluations.

INTERIM MANAGEMENT OF NUCLEAR MATERIALS

ENVIRONMENTAL IMPACT STATEMENT

STATEMENT FOR THE RECORD

This comment was given at the April 11, 1995, meeting in Savannah, Georgia:

I'd like to go on record as stating that the proposed action for the --for the Interim Nuclear Materials Siting Environmental Impact Statement is a -- not a concern but a --probably a better way for management and as a citizen of Georgia concerned that they are doing this and concerning -- and looking at these actions as opposed to the no-action alternative and that I support the proposed action alternative as a citizen of Georgia.

Karl D. Tesch
538 Forest Road
Hephzibah, GA 30815

Comment H4.

Response to oral comment H4:

DOE thanks the commentor for taking the time to comment on this EIS.

INTERIM MANAGEMENT OF NUCLEAR MATERIALS

ENVIRONMENTAL IMPACT STATEMENT

STATEMENT FOR THE RECORD

This comment was given at the April 13, 1995, meeting in North Augusta, South Carolina:

- Weapons grade plutonium-239 is extremely suitable for commercial reactors, and has an immense value for making electricity. H5-1
- A problem with DOE's approach to environmental impact statements is that it never conveys the overall picture. H5-2
- The only way to clean up Hanford Site and Idaho National Engineering Laboratory is to get rid of all the irradiated fuel. H5-3
- If DOE wants to provide more jobs at the Savannah River Site, it could process the Hanford Site's irradiated fuel at F-Canyon. H5-4
- When DOE talks about stabilizing its nuclear materials, it should realize that the materials are not chemically stabilized until they are converted to an oxide. H5-5
- Once materials are stabilized as oxides, DOE would have time to decide policy. H5-6

Robert F. Overman
724 Brucewood Street
Aiken, SC 29801

Comment H5 (summarization).

Response to oral comment H5:

H5-1

DOE acknowledges that plutonium could be used in reactors and has potential value in generating electricity. However, the civil use of plutonium for the generation of power is inconsistent with U.S. nonproliferation policy.

H5-2

The scope of this EIS is intended only to address the management of nuclear materials at the SRS. DOE believes it would be impractical to examine all actions that it might be contemplating nationwide in a single EIS. Where appropriate, the Department has prepared or is preparing EISs that cover national programs affecting multiple sites or areas. DOE has included the potential impacts from all actions that could involve the SRS in Chapter 5 of this EIS.

H5-3

Comment noted. However, the environmental restoration or cleanup of other DOE sites is outside the scope of this EIS.

H5-4

Comment noted. However, the management of the Hanford Reservation's irradiated fuel is outside the scope of this EIS.

H5-5

In the context of the EIS, the term "stable" describes a physical or chemical form of nuclear material that, when exposed to normal ambient air, does not undergo rapid change or degradation. DOE intends this term to convey to the reader that the material is "relatively stable" under normal storage conditions. DOE recognizes that for many materials to be "stable," extraordinary packaging or storage measures would be required (i.e., an inert atmosphere within an airtight container for plutonium metal). DOE acknowledges that only inert materials would classically be defined as stable. Again, DOE intends this term to imply slow physical or chemical changes over time that would not necessarily translate into a storage hazard. An example would be the eventual oxidation of unirradiated aluminum-clad fuel stored in borated concrete racks. The rate of oxidation is likely to be so slow that, for the purposes of the EIS, the material can conservatively be expected to be "stable" over the next 10 years.

H5-6

The comment is consistent with DOE's stated purpose and need to stabilize some nuclear materials at the SRS until the Department can make broader policy decisions on long-term management or disposition. However, DOE does not believe that it needs to convert all materials into an oxide for them to be considered "stable" (see the response to Comment H5-5 above). In fact, DOE's preferred alternatives would result in several physical forms (e.g., metal, oxide, glass) that would be stable.

INTERIM MANAGEMENT OF NUCLEAR MATERIALS

ENVIRONMENTAL IMPACT STATEMENT

STATEMENT FOR THE RECORD

This comment was given at the April 13, 1995, meeting in North Augusta, South Carolina:

H6-1 My comment has to do with what is -- what the EIS leads up to. I think the EIS is good. It's required by law; it gives alternatives, but the -- I think one thing that needs to be said needs to be said up front especially for these students here. The final decision on what -- how things are done or how things are evolved are really based on two facts: one is how much money DOE has to work with in the final decision and two is politics. Whether South Carolina voted for the current president and his cabinet or not has a very lot to do with it.

Perry Halton

Comment H6.

Response to oral comment H6:

DOE thanks the commentor for taking the time to comment on this EIS.

INTERIM MANAGEMENT OF NUCLEAR MATERIALS

ENVIRONMENTAL IMPACT STATEMENT

STATEMENT FOR THE RECORD

This comment was given at the April 13, 1995, meeting in North Augusta, South Carolina:

I think the aluminum-clad materials currently stored in the receiving basin for off-site fuel RBOF should be considered for the same treatment as the SRS reactor basin materials. The DNFSB recommended to the Secretary of Energy in Recommendation 94-1 that preparations be expedited to process deteriorating irradiated reactor fuel stored in basins at the Savannah River Site into a form suitable for safe interim storage until an option for ultimate disposition is selected. Aluminum-clad materials were not designed for indefinite storage. The fact that RBOF material currently shows no signs of deterioration does not preclude future deterioration or its inherent susceptibility.

I suggest that stabilization of RBOF-stored, aluminum-clad materials in the same manner as the reactor-basin materials should be considered for the following reasons: it will preclude future vulnerability possibilities; it will free more RBOF storage space for future receipt of other materials such as foreign research reactor fuel; stabilization by dissolution can eliminate the direct nuclear proliferation concerns if HE -- highly enriched uranium HEU material is isotopically dilute -- diluted to low-enriched uranium, LEU; recovered HEU when diluted to LEU has the possibility of being sold commercially, providing income for the U.S. Government; it can allow more economical use of taxpayer facilities by concurrent and consistent

Comment H7. Page 1 of 2.

treatment for similar materials preventing a proliferation of material forms; stabilization facility capabilities used for reactor materials may not exist after the ten-year stable material window.

I also think that unirradiated reactor component materials such as depleted uranium or neptunium targets should be able to be treated in the same manner as the irradiated material. This could allow for cold-run testing materials for stabilization processes and allow consolidation of similar types of materials. Economic benefits of reducing the variety of existing forms should be a consideration. And I'll give you a copy.

Charles R. Goergen
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Comment H7. Page 2 of 2.

Response to oral comment H7:

See responses to Comment letter L3.

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GLOSSARY

abnormal transients

A state resulting from an unusual incident in which operating parameters affecting control of radioactive materials move out of the normal operating range.

absorbed dose

The energy deposited per unit mass by ionizing radiation. The unit of absorbed dose is the rad.

air quality

A measure of the quantity of pollutants in the air.

air quality standards

The prescribed quantity of pollutants in the outside air that cannot be exceeded legally during a specified time in a specified area.

alpha (α) particle

A positively charged particle consisting of two protons and two neutrons that is emitted from the nucleus of certain nuclides during radioactive decay. It is the least penetrating of the three common types of radiation (alpha, beta, and gamma).

ambient air

The surrounding atmosphere, usually the outside air, as it exists around people, plants, and structures. It is not the air in immediate proximity to emission sources.

aqueous

In liquid form (i.e., dissolved in water).

aquifer

A geologic formation that contains sufficient saturated permeable material to conduct groundwater and to yield worthwhile quantities of groundwater to wells and springs.

atmosphere

The layer of air surrounding the earth.

AXAIR89Q

A computer model that is used to analyze doses from accidental airborne radionuclide releases. Developed in accordance with U.S. Nuclear Regulatory Commission Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accidental Consequence Assessments at Nuclear Power Plants*, February 1993.

background exposure

See exposure to radiation.

background radiation

Normal radiation present in the lower atmosphere from cosmic rays and earth sources. Background radiation varies with location, depending on altitude and natural radioactivity present in the surrounding geology.

beta (β) particle

An elementary particle emitted from a nucleus during radioactive decay. It is negatively charged, is identical to an electron, and is easily stopped by a thin sheet of metal.

bounded

TE | Producing greater or lesser consequences than any other accident (i.e., worst case or best case).

burial ground

A place for burying unwanted radioactive materials in which the earth acts to contain or prevent the escape of radiation. In this EIS, materials are incorporated into concrete to prevent the leaching of materials or movement in the underground environment.

button

Plutonium metal in a hemispherical shape, weighing about 1.8 kilograms (4 pounds).

°C

Degree Celsius. $°\text{C} = \frac{5}{9} \times (°\text{F} - 32)$.

cancer

A malignant tumor of potentially unlimited growth, capable of invading surrounding tissue or spreading to other parts of the body by metastasis.

canister

A stainless-steel container in which nuclear material is sealed.

canyon

A heavily shielded building used in the chemical processing of radioactive materials to recover special isotopes for national defense or other programmatic purposes. Operation and maintenance are by remote control.

capable (fault)

Determination if a geological fault has moved at or near the ground surface within the past 35,000 years.

carcinogen

An agent capable of producing or inducing cancer.

carcinogenic

Capable of producing or inducing cancer.

cask

A heavily shielded massive container for holding nuclear materials during shipment.

cesium

Naturally occurring element with 55 protons in its nucleus. Some manmade isotopes of cesium are radioactive (e.g., cesium 134, cesium-137).

cladding

The material (generally aluminum in SRS reactors) that covers each tubular fuel and target assembly.

collective dose

The sum of the individual doses to all members of a specific population.

committed effective dose equivalent

Used in cases when a person has an intake of radioactive material to denote that the dose is calculated for a period of 50 years following the intake. (See effective dose equivalent.)

community (environmental justice definition)

A group of people or a site within a spatial scope exposed to risks that potentially threaten health, ecology, or land values, or exposed to industry that stimulates unwanted noise, smell, industrial traffic, particulate matter, or other nonaesthetic impacts.

concentration

The amount of a substance contained in a unit quantity of a sample.

condensate

Liquid water obtained by cooling the steam (overheads) produced in an evaporator system.

constituents

Parts or components of a chemical system.

converting

The process for changing special isotopes into usable chemical forms to satisfy current or projected needs for a unique product.

criticality

A state in which a self-sustaining nuclear chain reaction is achieved.

cumulative effects

Additive environmental, health, and socioeconomic effects that result from a number of similar activities in an area.

curie (Ci)

A unit of radioactivity equal to 37,000,000,000 decays per second.

daughter

A nuclide formed by the radioactive decay of another nuclide, which is the "parent."

decay, radioactive

The spontaneous transformation of one nuclide into a different nuclide or into a different energy state of the same nuclide. The process results in the emission of nuclear radiation (alpha, beta, or gamma radiation).

decommissioning

The removal from service of facilities such as processing plants, waste tanks, and burial grounds, and the reduction or stabilization of radioactive contamination. Decommissioning concepts include:

- Decontaminate, dismantle, and return area to original condition without restrictions.
- Partially decontaminate, isolate remaining residues, and continue surveillance and restrictions.

defense waste

Nuclear waste generated by government defense programs as distinguished from waste generated by commercial and medical facilities.

depleted uranium

A mixture of uranium isotopes where uranium-235 represents less than 0.7 percent of the uranium by mass.

design-basis accident (DBA)

A postulated accident scenario for establishing the need for certain design features; normally, the accident that causes the most severe consequence when engineered safety features function as intended.

disposal/disposition

After designation as "surplus"; movement; placement in an onsite or offsite facility after a decision that future uses are unlikely or undesirable; determining whether the disposal of items must be "retrievable" under public law.

dose rate

The radiation dose delivered per unit time (e.g., rem per year).

ecology

The science dealing with the relationship of all living things with each other and with the environment.

ecosystem

A complex of the community of living things and the environment forming a functioning whole in nature.

effective dose equivalent

A quantity used to estimate the biological effect of ionizing radiation. It is the sum over all body tissues of the product of absorbed dose, the quality factor (to account for the different penetrating ability of the various radiations), and the tissue weighting factor (to account for the different radiosensitivity of the various tissues of the body).

effluent

Liquid or airborne material released to the environment. In general usage, however, "effluent" implies liquid releases.

effluent standards

Defined limits of effluent in terms of volume, content of contaminants, temperature, etc.

EIS

Environmental impact statement, a legal document required by the National Environmental Policy Act (NEPA) of 1969, as amended, for Federal actions involving potentially significant environmental impacts.

element

TC | One of the 109 known chemical substances that cannot be divided into simpler substances by chemical means. All isotopes of an element have the same atomic number (number of protons) but have a different number of neutrons.

Emergency Response Planning Guidelines (ERPG)

Values used to determine potential health effects from chemical accidents.

emission standards

Legally enforceable limits on the quantities and kinds of air contaminants that can be emitted into the atmosphere.

endangered species

Plants and animals in an area that are threatened with either extinction or serious depletion.

energy

The capacity to produce heat or do work.

environment

The sum of all external conditions and influences affecting the life, development, and ultimately the survival of an organism.

epicenter

The point on the earth's surface directly above the focus of an earthquake.

EPICODE

A computer model used to estimate the airborne concentration of toxic chemicals as a result of routine or accidental releases to the environment.

erosion

The process in which the actions of wind or water carry away soil and clay.

exceedence

A value over a prescribed limit.

exposure to radiation

The incidence of radiation on living or inanimate material by accident or intent. Background exposure is the exposure to natural background ionizing radiation. Occupational exposure is the exposure to ionizing radiation that occurs at a person's workplace. Population exposure is the exposure to a number of persons who inhabit an area.

°F

Degree Fahrenheit. $F = °C \times \frac{9}{5} + 32$.

fallout

The descent to earth and deposition on the ground of particulate matter (that might be radioactive) from the atmosphere.

fault

A fracture or a zone of fractures within a rock formation along which vertical, horizontal, or transverse slippage of the earth's crust has occurred in the past.

fissile

Capable of being split or divided (fissioned) by the absorption of thermal neutrons. The most common fissile materials are uranium-233, uranium-235, and plutonium-239.

fission

The splitting of a heavy nucleus into two approximately equal parts, which are nuclei of lighter elements, accompanied by the release of energy and generally one or more neutrons. Fission can occur spontaneously or can be induced by nuclear bombardment.

fission products

Nuclei from the fission of heavy elements (primary fission products); also, the nuclei formed by the decay of the primary fission products, many of which are radioactive.

floodplain

Level land built up by flowing stream deposition and periodically submerged by floodwater from that stream.

frit

Finely ground glass.

gamma (γ) rays

High-energy, short-wavelength electromagnetic radiation accompanying fission, radioactive decay, or nuclear reactions. Gamma rays are very penetrating and require relatively thick shields to absorb the rays effectively.

geology

The science that deals with the earth: the materials, processes, environments, and history of the planet, especially the lithosphere, including the rocks, their formation and structure.

glovebox

Large enclosure that separates workers from equipment used to process hazardous material but enables the workers to be in physical contact with the equipment; normally constructed of stainless steel with large acrylic/lead glass windows. Workers have access to equipment through the use of heavy-duty, lead-impregnated rubber gloves, the cuffs of which are sealed in portholes in the glovebox windows.

groundwater

The supply of fresh water under the earth's surface in an aquifer.

habitat

The place or type of site where a plant or animal naturally or normally lives and grows.

half-life (radiological)

The time in which half the atoms of a radioactive substance disintegrate to another nuclear form. Half-lives vary from millionths of a second to billions of years.

heavy metals

Metallic elements of high molecular weight, such as mercury, chromium, cadmium, lead, and arsenic, that are toxic to plants and animals at known concentrations.

HEPA filter

High efficiency particulate air filter designed to remove 99.9 percent of particles as small as 0.3 micrometer in diameter from a flowing air stream.

high-fired oxide

Oxide chemical form of plutonium produced by heating the material to approximately 1,000°C. High-fired oxide is considered more chemically stable than low-fired oxide because the higher heat removes moisture and other impurities more effectively.

high-level waste

The highly radioactive wastes that result from processing of defense materials at SRS.

historic resources

The sites, districts, structures, and objects considered limited and nonrenewable because of their association with historic events, persons, or social or historic movements.

immobilization

Conversion of high-level waste into a form that will be resistant to environmental dispersion.

ingot

A mass of metal cast into a convenient shape for storage or transportation. TE intensity (earthquake)

A numerical rating used to describe the effects of earthquake ground motion on people, structures, and the earth's surface. The numerical rating is based on an earthquake intensity scale such as the Richter Scale commonly used in the United States.

interim storage

Safe and secure capacity in the near term to support continuing operations in the interim period (10 years).

involved worker

TC For this EIS, an SRS worker who is involved in a given operation or activity when a radioactive release occurs.

ion

An atom or molecule that has gained or lost one or more electrons to become electrically charged.

ion exchange

Process in which a solution containing soluble ions to be removed is passed over a solid ion-exchange medium, which removes the soluble ions by exchanging them with labile ions from the surface of the column. The process is reversible so that the trapped ions can be collected (eluted) and the column regenerated.

ion-exchange medium

A substance (e.g., a resin) that preferentially removes certain ions from a solution.

ionization

The process that creates ions. Nuclear radiation, X-rays, high temperatures, and electric discharges can cause ionization.

ionizing radiation

TC Radiation capable of ejecting electrons from atoms or molecules to produce ions.

irradiation

Exposure to radiation.

ISC2

A computerized dispersion program used to calculate ground-level concentrations of air pollutants.

isotope

An atom of a chemical element with a specific atomic number and atomic weight. Isotopes of the same element have the same number of protons but different numbers of neutrons. Isotopes are identified by the name of the element and the total number of protons and neutrons in the nucleus. For example, plutonium-239 is a plutonium atom with 239 protons and neutrons.

LADTAP

A computer program used to calculate individual and population doses from liquid pathways.

latent cancer fatalities

Deaths resulting from cancer that has become active following a latent period (i.e., a period of inactivity).

low-fired oxide

Oxide chemical form of plutonium produced by heating the material to approximately 550°C. Low-fired oxide is considered less chemically stable than high-fired oxide because the lower heat does not remove moisture and other impurities as effectively.

low-income communities

A community where 25 percent or more of the population is identified as living in poverty.

low-level waste

Radioactive waste not classified as high-level waste; the wastes (mostly salts) remaining after removal of the highly radioactive nuclides from the liquid high-level wastes for immobilization.

Materials Test Reactor Fuel Equivalent (MTRFE)

A reactor fuel assembly that contains approximately the same volume as an assembly with the following physical parameters:

Maximum assembly unit weight 7.4 kilograms
Maximum assembly length 48.625 inches
Maximum assembly width 2.8 inches by 3.17 inches
Fuel clad thickness 0.015 inches

TC

Mark-x (Mk-x)

An historic naming system for a specific design of fuel or target material used in SRS production reactors (e.g., Mk-31, Mk-42, Mk-16).

MAXIGASP

A computer program used to calculate doses of airborne releases of radioactivity to the maximally exposed member of the public.

maximum contaminant levels (MCLs)

The maximum permissible level of a contaminant in water that is delivered to a user of a public water system.

maximally exposed individual

A hypothetical person located to receive the maximum possible dose by a given exposure scenario.

metric tons of heavy metal (MTHM)

TC Quantities of unirradiated and spent nuclear fuel and targets are traditionally expressed in terms of metric tons of heavy metal (typically uranium) without the inclusion of other materials such as cladding, alloy materials, and structural materials. A metric ton is 1,000 kilograms, which is equal to about 2,200 pounds.

migration

The natural travel of a material through the air, soil, or groundwater.

mitigate

To take practicable means to avoid or minimize environmental harm from a selected alternative.

monitoring

Continuing control and accountability, particularly of special nuclear materials such as plutonium-239 and highly enriched uranium, but also including oversight of hazardous or reactive compounds before they are disposed of or converted to a stable long-term storage form.

National Register of Historic Places

A list maintained by the National Park Service of architectural, historic, archaeological, and cultural sites of local, state, or national importance.

natural radiation or natural radioactivity

Background radiation. Some elements are naturally radioactive, whereas others are induced to become radioactive by bombardment in a reactor or accelerator.

NEPA

National Environmental Policy Act of 1969 (42 USC 4321); it requires the preparation of an EIS for Federal projects that could present significant impacts to the environment.

nonproliferation

The restriction of ability to easily access fissile materials in concentrations sufficient to assemble a nuclear weapon.

NO_x

Oxides of nitrogen, primarily nitrogen oxide (NO) and nitrogen dioxide (NO_2). These are produced in the combustion of fossil fuels, and can constitute an air pollution problem.

NRC

Nuclear Regulatory Commission; the independent Federal commission that licenses and regulates nuclear facilities.

nuclear energy

The energy liberated by a nuclear reactor (fission or fusion) or by radioactive decay.

nuclear radiation

Radiation, usually alpha, beta, or gamma, that emanates from an unstable atomic nucleus.

nuclear reaction

An interaction between a photon, particle, or nucleus and a target nucleus, leading to the emission of one or more particles and photons.

nuclear reactor

A device in which a fission chain reaction is maintained, used for the irradiation of materials or the generation of electricity.

nuclide

An atomic nucleus specified by atomic weight, atomic number, and energy state; a radionuclide is a radioactive nuclide.

organic compounds

Chemical compounds containing carbon.

outfall

Place where liquid effluents enter the environment and are monitored.

oxide

TC A compound in which an element is chemically combined with oxygen.

ozone

A compound of oxygen in which three oxygen atoms are chemically attached to each other.

particulates

Solid particles and liquid droplets small enough to become airborne.

passive safety system

A system that provides safety features requiring no human intervention or adverse condition to actuate.

passivation

TC The process of making metals inactive or less chemically reactive (for example, to passivate the surface of steel by chemical treatment).

pH

A measure of the hydrogen ion concentration in aqueous solution. Pure water has a pH of 7, acidic solutions have a pH less than 7, and basic solutions have a pH greater than 7.

people of color communities

A population classified by the U.S. Bureau of the Census as Black, Hispanic, Asian and Pacific Islander, American Indian, Eskimo, Aleut, and other nonwhite persons, the composition of which is at least equal to or greater than the state minority average of a defined area or jurisdiction.

permeability

Ability of liquid to flow through rock, groundwater, soil, or other substance.

person-rem

The radiation dose to a given population; the sum of the individual doses received by a population segment.

physiographic

Geographic regions based on geologic setting.

plutonium (Pu)

A transuranic, heavy (average atomic mass about 244 atomic mass units), silvery metal with 15 known isotopes that is produced by the neutron irradiation of natural uranium. TE Plutonium-239 is used both in nuclear weapons and commercial nuclear power applications. Plutonium-238 is used to power onboard generators during manned and unmanned space flights.

plutonium solutions

Chemical solutions containing plutonium.

poison

A material that has an affinity for absorbing neutrons. Poisons are added to nuclear materials with a potential criticality concern to lessen the likelihood of an uncontrolled nuclear reaction.

pollution

The addition of an undesirable agent to an ecosystem in excess of the rate at which natural processes can degrade, assimilate, or disperse it.

POPGASP

A computer mathematical model used to calculate doses of airborne releases of radioactivity to the population within 80 kilometers (50 miles) of the SRS.

population

In this EIS, a collection of members of the public who are located outside the boundaries of the SRS. Impacts in this EIS are estimated for the population within a given area, depending on the appropriate environmental pathways. For example, the affected population for liquid releases to the Savannah River includes downstream residents.

precipitate

A solid (used as a noun).

To form a solid substance in a solution by a chemical reaction (used as a verb).

PUREX process

A chemical separation process to retrieve plutonium, uranium, and other radionuclides from reactor fuel and targets.

radiation

The emitted particles and photons from the nuclei of radioactive atoms; a shortened term for ionizing radiation or nuclear radiation as distinguished from nonionizing radiation (microwaves, ultraviolet rays, etc.).

radioactivity

The spontaneous decay of unstable atomic nuclei, accompanied by the emission of radiation.

radioisotopes

Radioactive isotopes. Some radioisotopes are naturally occurring (e.g., potassium-40) while others are produced by nuclear reactions.

radiolysis

TC The decomposition of a material (usually water) into different molecules due to ionizing radiation. In water, radiolysis results in the production of such molecules as hydrogen gas, oxygen, and hydrogen peroxide.

repository

A place for the disposal of immobilized high-level waste in isolation from the environment.

resin

An ion-exchange medium; organic polymer used for the preferential removal of certain ions from a solution.

Richter Scale

A scale of measure used in the United States to quantify earthquake intensity.

risk

In accident analysis, the probability weighted consequence of an accident, defined as the accident frequency per year multiplied by the dose. The term "risk" is also used commonly in other applications to describe the probability of an event occurring.

runoff

The portion of rainfall, melted snow, or irrigation water that flows across ground surface and eventually returns to streams. Runoff can carry pollutants into receiving waters.

saltstone

Low-radioactivity fraction of high-level waste from the in-tank precipitation process mixed with cement, flyash, and slag to form a concrete block.

seismicity

The tendency for earthquakes to occur.

shield

Material used to reduce the intensity of radiation that would irradiate personnel or equipment.

short-lived

A designation for radionuclides with relatively short half-lives (i.e., they decay to other atoms relatively quickly). | TC

stabilization

The action of making a nuclear material more stable by converting its physical or chemical form or placing it in a more stable environment.

stack

A vertical pipe or flue designed to exhaust gases and suspended particulates.

strontium

Naturally occurring element with 38 protons in its nucleus. Some manmade isotopes of strontium are radioactive (e.g., strontium-89, strontium-90),

surface water

All water on the surface (streams, ponds, etc.), as distinguished from underground water.

tank farm

An installation of interconnected underground tanks for the storage of high-level radioactive liquid wastes.

target

In this EIS, a tube of material placed in a reactor to absorb neutrons and be changed to a desired end product.

transuranic waste

Waste material containing more than a specified concentration of transuranic elements (presently, more than 10 nanocuries per gram of waste).

tritium

A radioactive isotope of hydrogen; its nucleus contains one proton and two neutrons.

uninvolved worker

TC For this EIS, an SRS worker who is not involved in a given operation or activity when a radioactive release occurred.

uranium (U)

A heavy (average atomic mass of about 238 atomic mass units), silvery-white metal with 14 radioactive isotopes. One of the isotopes, uranium-235, is most commonly used as fuel for nuclear fission and another, uranium-238, is transformed into fissionable plutonium-239 following its capture of a neutron in a nuclear reactor.

vault

A reinforced concrete structure for storing strategic nuclear materials used in national defense or other programmatic purposes.

vitrification

Incorporation of a material into a glass form.

vulnerability

Condition or weakness that could lead to exposure to the public, unnecessary or increased exposure to workers, or release of radioactive materials to the environment.

waste, radioactive

Materials from nuclear operations that are radioactive or are contaminated with radioactive materials and for which there is no practical use or for which recovery is impractical.

ACRONYMS AND ABBREVIATIONS

Acronyms

AEC	Atomic Energy Commission
ALARA	as low as reasonably achievable
Am	americium
AQCR	Air Quality Control Region
ATTA	Advanced Tactical Training Area
C	carbon
CAA	Clean Air Act
Ce	cerium
CEQ	Council on Environmental Quality
CFR	Code of Federal Regulations
Cm	curium
Co	cobalt
Cs	cesium
CSEG	Criticality Safety Evaluation Group
CWA	Clean Water Act
D&D	decontamination and decommissioning
DNFSB	Defense Nuclear Facilities Safety Board
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
DU	depleted uranium
DWPF	Defense Waste Processing Facility
EA	environmental assessment
EIS	environmental impact statement
EPA	U.S. Environmental Protection Agency

TE

	ERPG	Emergency Response Planning Guideline
	ES&H	Environment, Safety and Health
	Eu	europium
	FFA	Federal Facility Agreement
	FFCA	Federal Facilities Compliance Agreement
	FR	Federal Register
	FWS	U.S. Fish and Wildlife Service
	H	hydrogen
	H-3	tritium
	HEPA	high-efficiency particulate air (filter)
	HEU	highly enriched uranium
	HLW	high-level waste
	I	iodine
	IAEA	International Atomic Energy Agency
	ICRP	International Commission on Radiological Protection
	IMNM	Interim Management of Nuclear Materials
	lcf	latent cancer fatality
	LEU	low enriched uranium
	MEI	maximally exposed individual
	mrem	millirem (1/1000 rem)
TE	MTHM	metric tons of heavy metal
	MTRFE	Materials Test Reactor Fuel Equivalent
	NAAQS	National Ambient Air Quality Standard
	NASA	National Aeronautics and Space Administration
	NCRP	National Council on Radiation Protection and Measurements
	NCSE	Nuclear Criticality Safety Evaluations
	NEPA	National Environmental Policy Act

Ni	nickel
NIM	Nuclear Incident Monitor
Np	neptunium
NPDES	National Pollutant Discharge Elimination System
OSHA	Occupational Safety and Health Administration
PEL	Permissible Exposure Limit
Pu	plutonium
RBOF	Receiving Basin for Offsite Fuel
RCRA	Resource Conservation and Recovery Act
rem	roentgen equivalent man
RINM	reactor irradiated nuclear materials
ROD	Record of Decision
ROI	region of influence
Ru	ruthenium
S	sulfur
SAR	Safety Analysis Report
Sb	stibium (antimony)
SCDHEC	South Carolina Department of Health and Environmental Control
SNF	spent nuclear fuel
Sr	strontium
SREL	Savannah River Ecology Laboratory
SRS	Savannah River Site
SRTC	Savannah River Technology Center
TC	Technical correction or change
TE	Technical editorial correction or change
TRU	transuranic
U	uranium

TE

	UNH	uranyl nitrate hexahydrate
	VOC	volatile organic compound
	USF	Uranium Solidification Facility
	WSRC	Westinghouse Savannah River Company
	Xe	xenon
	Zr	zirconium

Abbreviations for measurements

	cfm	cubic feet per minute
	cfs	cubic feet per second
	g	acceleration due to gravity (seismology)
	g/L	grams per liter
	gpm	gallons per minute
	L	liter
	lb	pound
	km	kilometer
	m	meter
	mg	milligram
	mo	month
TE	MW	megawatt
	pCi	picocurie
	yr	year
	µCi	microcurie
	µg	microgram
TE	µm	micron (micrometer)

°C	degrees Celsius
°F	degrees Fahrenheit

METRIC SYSTEM

Typically, scientific reports use metric units; therefore, this EIS presents metric units of measure (meters, liters, grams, etc.) rather than the more common U.S. Customary Units (feet, gallons, pounds, etc.). However, the text sections also provide U.S. Customary Units in parentheses for ease of understanding.

Many metric measurements presented include prefixes that denote a multiplication factor that is applied to the base standard (e.g., 1 kilometer = 1,000 meters). The following list presents these metric prefixes and applicable symbols:

TE

mega (M)	1,000,000 (10^6; one million)
kilo (k)	1,000 (10^3; one thousand)
hecto (h)	100 (10^2; one hundred)
centi (c)	0.01 (10^{-2}; one one-hundredth)
milli (m)	0.001 (10^{-3}; one one-thousandth)
micro (µ)	0.000001 (10^{-6}; one one-millionth)
nano (n)	0.000000001 (10^{-9}; one one-billionth)
pico (p)	0.000000000001 (10^{-12}; one one-quadrillionth)

DOE Order 5900.2A, "Use of the Metric System of Measurement" (3/26/92) prescribes the use of this system in DOE documents. The following list presents conversion factors for the metric units used in this EIS as an aid to readers who are more familiar with U.S. Customary Units.

Conversion from Metric to U.S. Customary Units

1 meter = 3.281 feet = 39.37 inches = 1.094 yards

1 kilometer = 0.6214 mile

1 square meter = 10.764 square feet

1 square kilometer = 0.3861 square mile = 247.1 acres

1 liter = 61.025 cubic inches = 0.2642 gallon

1 cubic meter = 35.31 cubic feet = 1.308 cubic yards = 264.2 gallons

Discharge

1 cubic meter per second = 35.31 cubic feet per second = 15,850.3 gallons per minute

Mass

1 kilogram = 2.205 pounds (mass)

1 metric ton = 2,205 pounds = 1.1023 (short) tons

Pressure

1 Pascal = 0.02089 pound per square foot

1 kilogram (force) per square meter = 0.2048 pound (force) per square foot

Velocity

1 meter per second = 3.281 feet per second = 2.237 miles per hour

1 kilometer per hour = 0.6214 mile per hour

Temperature

°C to °F, °C × 1.8 + 32 (i.e., 20°C = 20 × 1.8 + 32 = 68°F)

DISTRIBUTION LIST

DOE is providing copies of the final EIS to Federal, state, and local elected and appointed officials and agencies of government; Native American groups; Federal, state, and local environmental and public interest groups; and other organizations and individuals listed below. Copies will be provided to other interested parties upon request.

CONTENTS

Section — Page

A. United States Congress DL-2
  - A.1 Senators from Affected and Adjoining States DL-2
  - A.2 United States Senate Committees DL-2
  - A.3 United States House of Representatives from Affected and Adjoining States DL-2
  - A.4 United States House of Representatives Committees DL-3

B. Federal Agencies DL-3

C. State of South Carolina DL-6
  - C.1 Statewide Offices and Legislature DL-6
  - C.2 State and Local Agencies and Officials DL-7

D. State of Georgia DL-8
  - D.1 Statewide Offices and Legislature DL-8
  - D.2 State and Local Agencies and Officials DL-9

E. Native American Groups DL-9

F. Citizens Advisory Board Members DL-10

G. Environmental and Public Interest Groups DL-10
  - G.1 National DL-10
  - G.2 State and Local DL-12

H. Other Groups and Individuals DL-12

I. Reading Rooms DL-20

A. UNITED STATES CONGRESS

A.1 Senators from Affected and Adjoining States

The Honorable Paul Coverdell
United States Senate

The Honorable Ernest F. Hollings
United States Senate

The Honorable Sam Nunn
United States Senate

The Honorable Strom Thurmond
United States Senate

A.2 United States Senate Committees

The Honorable Strom Thurmond
Chairman
Committee on Armed Services

The Honorable Mark O. Hatfield
Chairman
Committee on Appropriations

The Honorable Trent Lott
Chairman
Subcommittee on Strategic Forces
Committee on Armed Services

The Honorable Pete V. Domenici
Chairman
Subcommittee on Energy and Water
Development
Committee on Appropriations

The Honorable Sam Nunn
Ranking Minority Member
Committee on Armed Services

The Honorable Robert C. Byrd
Ranking Minority Member
Committee on Appropriations

The Honorable J. James Exon
Ranking Minority Member
Subcommittee on Strategic Forces
Committee on Armed Services

The Honorable J. Bennett Johnston
Ranking Minority Member
Subcommittee on Energy and Water
Development
Committee on Appropriations

A.3 United States House of Representatives from Affected and Adjoining States

The Honorable James E. Clyburn
U.S. House of Representatives

The Honorable Nathan Deal
U.S. House of Representatives

The Honorable Lindsey Graham
U.S. House of Representatives

The Honorable Jack Kingston
U.S. House of Representatives

The Honorable Cynthia McKinney
U.S. House of Representatives

The Honorable Charlie Norwood
U.S. House of Representatives

The Honorable Mark Sanford
U.S. House of Representatives

The Honorable Floyd Spence
U.S. House of Representatives

The Honorable John M. Spratt, Jr.
U.S. House of Representatives

A.4 United States House of Representatives Committees

The Honorable Floyd Spence
Chairman
Committee on National Security

The Honorable Bob Livingston
Chairman
Committee on Appropriations

The Honorable Duncan Hunter
Chairman
Subcommittee on Military Procurement
Committee on National Security

The Honorable John T. Myers
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Subcommittee on Energy and Water Development
Committee on Appropriations

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Committee on National Security

The Honorable David R. Obey
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The Honorable Ike Skelton
Ranking Minority Member
Subcommittee on Military Procurement
Committee on National Security

The Honorable Tom Bevill
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Committee on Appropriations

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C. STATE OF SOUTH CAROLINA

C.1 Statewide Offices and Legislature

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The Honorable Charles Condon
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The Honorable Greg Ryberg
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The Honorable Rudy M. Mason
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The Honorable Charles Sharpe
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The Honorable Joseph P. Wilder
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The Honorable Thomas L. Moore
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Mayor of Hilton Head Island

The Honorable Thomas W. Greene
Mayor of North Augusta

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South Carolina Department of Archives and History

Commissioner
South Carolina Department of Health and Environmental Control

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South Carolina Department of Health and Environmental Control

Chief
Bureau of Drinking Water Protection
South Carolina Department of Health and Environmental Control

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South Carolina Department of Health and Environmental Control

Chief
Bureau of Radiological Health
South Carolina Department of Health and Environmental Control

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Bureau of Solid and Hazardous Waste Management
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South Carolina Geological Survey

Chairman of the Board
Beaufort-Jasper Water & Sewer Authority

Director
South Carolina State Development Board

Executive Director
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INDEX

Accidents S-4, S-12, 1-8, 1-9, 1-19, 1-20, 1-26, 2-5, 2-47, 2-59, 2-82, 2-83, 2-92, 2-93, 2-94, 2-95, 2-96, 2-97, 2-101, 2-102, 4-2, 4-3, 4-17, 4-41, 4-63, 4-64, 4-65, 4-66, 4-67, 4-68, 4-69, 4-70, 4-71, 4-72, 4-73, 4-76, 4-77, C-38, E-1, E-2, E-3, E-5, E-7, E-8, E-9, E-10, E-11, E-12, E-13, E-55, E-56, E-57, E-58, E-59, E-60, E-61, E-64, E-66, E-67, E-68, E-70, E-71, E-72, E-73, E-74, E-75, E-76

Acid Recovery Unit C-24, C-26

Actinide Packaging Facility S-9, S-10, 2-6, 2-15, 2-18, 2-19, 2-24, 2-28, 2-29, 2-30, 2-32, 2-33, 2-42, 2-43, 2-64, 2-73, 2-74, 2-76, 2-79, 2-80, 2-88, 2-89, 7-1, D-1

Aesthetics and Scenic Resources S-13, 4-3, 4-79

Aging facilities S-13, S-14, 2-81

Air emissions 4-33, 8-2

Air quality 2-63, 3-13, 3-14, 3-16, 3-17, 3-25, 4-46, 4-79, 4-80, 5-6, 5-7, 8-2

Aluminum-clad fuel 1-17, 2-34, 2-68, 5-2, A-2, E-58

Aluminum-clad targets S-3, S-6, S-7, S-9, 1-24, 2-31, 2-40, 2-41, 2-68, 2-101, 4-33, 4-48, 4-51, 4-55, 4-61, 5-1, 5-2, A-13

Americium S-6, S-10, 1-19, 1-20, 1-22, 1-23, 2-5, 2-20, 2-21, 4-71, A-6, C-46

Americium-241 S-1, 1-1, 5-4, A-6, A-9, B-4, E-6, E-60

Americium and curium S-2, S-3, S-6, S-7, 1-10, 1-20, 1-23, 1-24, 2-5, 2-11, 2-13, 2-14, 2-15, 2-21, 2-64, 2-68, 2-71, 2-77, 2-79, 2-91, 2-93, 2-94, 2-96, 4-1, 4-31, 4-40, 4-47, 4-49, 4-54, 4-57, 5-2, A-6, C-48, E-60

AXAIR89Q 4-76, E-7

Blending down S-9, S-10, 1-27, 2-25, 2-37, 2-61, 2-64, 2-65, 2-66, 2-68, 2-95, 2-99, 2-100, 4-31, 4-32, 4-47, 4-48, 4-50, 4-55, 4-59, 5-2, 5-11

Borosilicate glass S-10, C-49, C-56

Californium-252 S-1, 1-1, 1-10, 1-16, 1-23, 2-5, 2-11, C-30, C-46

Cancer risk 4-18, 5-14, 5-15, E-2

Candidate materials for stabilization 2-17, 2-83, A-8

Capable 1-5, 3-4, 3-7, 4-76

Cation exchange C-17, C-18, C-20

Cesium-137 1-15, 3-10, 3-11, 4-52, 5-15, B-4

Chem-Nuclear Services facility 5-10

Committed effective dose equivalent 3-22, B-4

Continuing Storage Alternative 2-60, 2-80, 2-81, 2-101, 4-30, 4-31, 4-40, 4-48, 4-49, 4-50, 4-54, 4-55, 4-56, 4-57, 4-58, 4-59, 4-60, 4-61, 4-62, 4-63, 4-65, 4-70

Core competency S-13, 2-80, 2-81

Criticality S-5, 1-14, 1-15, 1-16, 1-20, 1-22, 1-26, 2-20, 2-21, 2-26, 2-28, 2-29, 2-31, 2-36, 2-39, 2-40, 2-45, 2-73, 2-100, 2-105, 4-62, 4-65, A-1, B-3, B-4, C-28, C-36, E-55, E-56, E-57, E-73, E-74, E-75, E-76, E-78

Cultural resources S-13, 3-1, 4-3, 4-78

Cumulative impacts 1-26, 1-27, 1-28, 1-29, 1-30, 2-67, 5-1, 5-10, 5-13

Curium S-6, 1-19, 1-22, 1-23, 2-5, 2-20, 2-21, 4-71, 5-4, A-4, A-6, B-4, C-46

Decontamination and decommissioning 1-13, 1-19, 1-28, 2-83, 2-89, 5-8, 5-9, 5-14, C-1

Defense Nuclear Facilities Safety Board S-2, S-5, 1-9, 1-22, 2-3, 2-90, 2-91, 2-107

Defense Waste Processing Facility (DWPF) S-3, S-10, S-12, 1-29, 2-3, 2-6, 2-8, 2-10, 2-14, 2-17, 2-19, 2-21, 2-22, 2-26, 2-28, 2-29, 2-30, 2-31, 2-35, 2-36, 2-37, 2-39, 2-40, 2-45, 2-59, 2-60, 2-61, 2-62, 2-65, 2-66, 2-68, 2-80, 2-81, 2-90, 2-92, 2-96, 2-97, 2-99, 2-100, 2-102, 2-106, 4-1, 4-30, 4-31, 4-32, 4-33, 4-47, 4-48, 4-49, 4-50, 4-51, 4-54, 4-55, 4-56, 4-57, 4-58, 4-59, 4-60, 4-61, 4-62, 4-63, 5-2, 5-3, 5-9, 5-10, 5-13, 5-14, 5-16, 5-17, 6-1, C-5, C-30, C-49, C-56, E-8, E-58, E-59, E-61, E-68, E-70

Dry Storage Facility S-10, S-11, S-12, 2-34, 2-35, 2-38, 2-39, 2-40, 2-41, 2-44, 2-45, 2-64, 2-76, 2-77, 2-88, 2-98, 2-100, 2-106, 3-1, 4-2, 4-32, 4-33, 4-48, 4-60, 4-61, 4-62, 4-77, 4-78, 4-79, 4-80, 4-81, 5-1, 5-4, 7-1, C-32, C-34, C-36, E-58

Earthquakes S-1, 1-5, 2-81, 3-6, 3-7, 4-64, C-32, C-38, E-1, E-5, E-72

Ecology 4-81

Effective dose equivalent 3-22, 3-25, 3-26

Effluent Treatment Facility 1-28, C-10, C-24, C-49, C-56, C-67

Employment S-13, 3-18, 4-78, 8-7, 8-8

Environmental justice 3-19, 3-20, 4-2, 4-41, 4-46, 4-77, 8-9, E-71

Environmental Protection Agency S-1, 3-16, 3-19, 4-70, 5-6, 8-2, 8-3, 8-4, 8-5, 8-7, C-53, E-61

F-Area S-7, S-10, S-12, S-13, S-9, 1-24, 1-28, 2-8, 2-9, 2-10, 2-11, 2-13, 2-14, 2-15, 2-16, 2-18, 2-20, 2-23, 2-24, 2-25, 2-26, 2-28, 2-29, 2-30, 2-31, 2-32, 2-33, 2-36, 2-38, 2-42, 2-43, 2-45, 2-46, 2-61, 2-62, 2-64, 2-71, 2-73, 2-74, 2-79, 2-80, 2-107, 3-1, 3-4, 3-8, 3-10, 3-11, 4-2, 4-3, 4-47, 4-51, 4-53, 5-9, 7-1, A-6, A-8, C-5, C-13, C-14, C-16, C-24, C-27, C-31, C-42, C-49, C-53, C-56, C-67, E-7, E-8, E-58, E-61, E-62, E-63, E-64, E-68

F-Canyon S-1, S-7, S-9, S-10, 1-5, 1-8, 1-10, 1-11, 1-15, 1-17, 1-18, 1-20, 1-21, 1-22, 1-24, 1-25, 1-26, 1-27, 2-5, 2-8, 2-9, 2-11, 2-13, 2-14, 2-15, 2-16, 2-20, 2-23, 2-24, 2-26, 2-28, 2-30, 2-31, 2-32, 2-33, 2-36, 2-37, 2-42, 2-43, 2-44, 2-45, 2-46, 2-60, 2-61, 2-64, 2-65, 2-66, 2-68, 2-71, 2-74, 2-76, 2-77, 2-79, 2-80, 2-89, 2-91, 2-93, 2-96, 2-97, 2-98, 2-103, 2-107, 3-10, 3-27, 3-28, 4-17, 4-30, 4-31, 4-32, 4-40, 4-47, 4-48, 4-57, 4-58, 4-60, 4-62, 5-2, 5-11, 5-13, 5-14, 5-15, 5-16, 5-17, 6-1, 7-1, A-6, A-8, A-12, C-5, C-9, C-10, C-12, C-13, C-17, C-27, C-46, C-47, C-48, C-65, E-3, E-5, E-6, E-55

F-Canyon Plutonium Solutions EIS 1-25, 1-26, 5-16

F-Canyon Vitrification Facility 2-8, 2-13, 2-31, 2-36, 2-46, 2-64, 2-65, C-48

FA-Line Facility C-13, E-55, E-56

FB-Line S-7, S-9, S-10, 1-11, 1-20, 1-24, 2-6, 2-9, 2-10, 2-16, 2-18, 2-19, 2-23, 2-24, 2-27, 2-28, 2-29, 2-30, 2-31, 2-32, 2-33, 2-34, 2-36, 2-42, 2-43, 2-44, 2-45, 2-65, 2-71, 2-73, 2-74, 2-76, 2-77, 2-79, 2-98, 2-104, 4-31, 4-57, 4-58, 4-59, 4-61, 4-62, A-6, A-9, A-10, A-11, A-12, C-13, C-14, C-17, C-18, C-20, C-21, C-45, E-56

Fissile material S-6, S-10, 1-9, 1-15, 1-16, 2-20, 2-21, 2-22, 2-23, 2-27, 2-28, 2-29, 2-31, 2-37, 2-40, 2-60, 2-73, 2-91, 2-100, 2-102, 2-105, 2-106, A-1, A-8, A-10, E-74, E-75

GASPAR 4-46

Glovebox 2-10, 2-18, 2-24, 2-27, 2-28, 2-29, 2-32, 2-42, 2-68, A-8, C-22, C-33, C-34, C-45, C-46, E-5, E-56, E-57, E-60, E-77

Groundwater 2-81, 3-8, 3-10, 3-11, 3-12, 3-25, 4-52, 5-9, 5-10, 5-16, 5-17, A-6

H-Area S-7, S-10, S-13, 1-24, 1-28, 2-6, 2-18, 2-19, 2-20, 2-25, 2-26, 2-29, 2-30, 2-37, 2-61, 2-62, 2-64, 2-71, 2-73, 2-76, 2-79, 2-80, 2-95, 2-107, 3-1, 3-4, 3-8, 3-10, 3-11, 4-2, 4-3, 4-47, 4-51, 4-53, 4-54, 4-59, 4-81, 5-9, 7-1, A-8, A-12, C-5, C-16, C-17, C-24, C-28, C-49, C-53, C-56, C-61, C-67, E-57, E-58, E-62, E-64, E-65, E-66, E-68

H-Canyon S-1, S-7, S-10, 1-5, 1-8, 1-15, 1-17, 1-18, 1-20, 1-21, 1-22, 1-24, 1-27, 1-32, 2-5, 2-6, 2-8, 2-9, 2-10, 2-15, 2-16, 2-17, 2-18, 2-19, 2-20, 2-23, 2-24, 2-25, 2-26, 2-28, 2-29, 2-30, 2-33, 2-36, 2-37, 2-38, 2-39, 2-44, 2-45, 2-61, 2-63, 2-64, 2-65, 2-66, 2-68, 2-73, 2-74, 2-76, 2-79, 2-80, 2-81, 2-89, 2-94, 2-95, 2-107, 3-10, 3-11, 3-27, 3-28, 4-1, 4-31, 4-40, 4-47, 4-48, 4-50, 4-55, 4-58, 4-59, 4-65, 5-2, 5-4, 6-1, A-8, C-5, C-9, C-14, C-15, C-16, C-21, C-23, C-28, C-32, C-33, C-56, C-65, E-5, E-6, E-55

Health effects S-3, 2-47, 2-63, 3-22, 3-27, 4-2, 4-17, 4-18, 4-30, 4-31, 4-32, 4-33, 4-34, 4-39, 4-41, 4-46, 4-52, 4-64, 4-65, 4-71, 4-76, 5-5, 5-6, 5-11, 5-12, 5-14, 5-15, B-5, E-9, E-10, E-11, E-55, E-57, E-64, E-66, E-67

HEPA filter C-34, C-48

High efficiency particulate air filter 1-5, C-7, C-34

High-level waste S-10, S-12, 1-13, 1-19, 1-28, 2-3, 2-6, 2-9, 2-10, 2-13, 2-14, 2-15, 2-16, 2-17, 2-18, 2-1-9, 2-20, 2-21, 2-23, 2-24, 2-25, 2-26, 2-28, 2-29, 2-30, 2-31, 2-33, 2-36, 2-37, 2-38, 2-39, 2-40, 2-43, 2-45, 2-46, 2-60, 2-65, 2-83, 2-90, 2-96, 2-102, 4-56, 4-57, 4-58, 4-59, 4-60, 4-61, 4-62, 5-9, 5-16, 6-1, C-12, C-13, C-14, C-16, C-17, C-27, C-49, C-56, E-8, E-58

Highly enriched uranium S-7, S-9, S-10, S-14, 1-22, 1-23, 1-24, 1-26, 1-27, 1-28, 1-30, 1-31, 2-4, 2-25, 2-26, 2-29, 2-37, 2-39, 2-64, 2-65, 2-66, 2-71, 2-73, 2-76, 2-79, 2-105, 5-11, 5-13, 5-14, 5-15, 5-16, 5-17, A-8, A-13, A-14, A-15, C-5, C-14, C-15, C-16, C-32, C-33

Hot canyon 1-5, 2-11, C-9, C-12, C-13, C-15, C-16, C-17, C-24, C-26, C-47

Improving Storage Alternative S-12, 2-41, 2-60, 2-64, 2-81, 2-83, 2-92, 2-96, 2-97, 2-99, 2-100, 2-101, 2-102, 2-104, 4-32, 4-33, 4-40, 4-48, 4-49, 4-55, 4-56, 4-60, 4-61, 4-62, 5-1

Ion exchange C-67

Irreversible and irretrievable commitments of resources 7-1

K-Reactor S-7, 1-15, 1-16, 1-24, 2-37, 2-74, 2-76, 3-4, A-7, C-1, C-6

K-Reactor Disassembly Basin 2-31, A-12, A-13

LADTAP II 4-52

Land use 4-3, 4-78

Latent cancer fatalities S-11, S-12, 2-47, 2-59, 4-17, 4-18, 4-30, 4-34, 4-35, 4-36, 4-37, 4-38, 4-39, 4-40, 4-41, 4-64, 4-65, 4-66, 4-67, 4-68, 4-69, 4-70, 4-71, 4-72, 4-73, 4-74, 4-75, 4-76, 5-5, 5-14, 5-15, E-2, E-8, E-9, E-60

Latent cancer fatality 2-47, 4-17, 4-18, 4-39, 4-41, 4-64, 4-65, 4-76, 5-13, 5-14, 5-16, E-55, E-59, E-60

Low enriched uranium S-9, S-10, 1-27, 2-25, 2-26, 2-37, 2-61, 2-64, 2-65, 2-66, 2-68, 2-71, 2-73, 2-79, 2-80, 2-95, 2-99, 2-100, 2-105, 4-31, 4-32, 4-47, 4-48, 4-50, 4-55, 4-59, 5-2, A-13, A-14, C-49, E-55

Low-level waste 1-13, 1-9, 2-62, 2-90, 2-96, 3-25, 4-34, 4-56, 4-57, 4-58, 4-59, 4-60, 4-61, 4-62, 4-63, 5-3, 5-8, 5-9, 5-10, 5-16, 5-13, 5-16, C-27, C-59, C-61

Mark-16 fuels S-3, S-7, S-9, S-10, 1-16, 1-17, 1-24, 4-32, 4-40, 4-48, 4-50, 4-55, 4-61, 5-2, A-12

Mark-22 fuel S-3, S-7, S-9, S-10, 1-15, 1-16, 1-17, 1-24, 2-37, 2-38, 2-39, 2-41, 2-45, 2-60, 2-64, 2-68, 2-99, 2-104, 2-105, 2-106, 4-32, 4-40, 4-48, 4-50, 4-55, 4-61, 5-2, A-7, A-13

Mark-31 targets S-3, S-7, S-9, 1-11, 1-15, 1-16, 1-17, 1-18, 1-24, 1-25, 1-26, 2-31, 2-33, 2-34, 2-35, 2-36, 2-37, 2-38, 2-39, 2-41, 2-45, 2-60, 2-63, 2-64, 2-66, 2-68, 2-71, 2-74, 2-77, 2-79, 2-88, 2-92, 2-97, 2-98, 2-99, 2-103, 2-104, 2-105, 2-106, 4-32, 4-33, 4-34, 4-48, 4-50, 4-55, 4-60, 5-1, 5-2, A-7, A-12, E-5

MAXIGASP 4-46

Maximally exposed individual 2-61, 2-62, 3-25, 4-17, 4-18, 4-41, 4-43, 4-46, 4-52, 4-71, 4-76, 5-5, 5-6, 5-7, 5-11, 5-13, 5-14, 5-15, E-59, E-60, E-62, E-64, E-72

Mixed waste 1-13, 1-28, 2-62, 2-90, 2-96, 4-34, 4-56, 4-60, 4-61, 4-62, 5-3, 5-8, 5-9, 5-16, 8-5, C-49, C-53, C-56, C-65, C-67

National Environmental Policy Act S-9, 1-18, 1-25, 1-31, 2-8, 2-77, 2-99, 2-103, 2-104, 2-105, 4-34, 5-10, 8-1, 8-11, C-46

National Register of Historic Places 3-1, 4-78

Neptunium S-3, S-6, S-7, S-9, 1-20, 1-22, 1-23, 1-24, 2-5, 2-15, 2-16, 2-17, 2-20, 2-21, 2-27, 2-37, 2-64, 2-68, 2-73, 2-74, 2-76, 2-79, 2-80, 2-93, 4-1, 4-31, 4-34, 4-40, 4-41, 4-47, 4-50, 4-54, 4-58, 5-2, 5-4, A-8, A-11, B-4, C-1, C-5, C-12, C-14, C-15, C-16, C-17, C-22, C-23, C-31, E-60, E-68

No-Action Alternative 2-1, 2-3, 2-61, 2-67, 2-69, 2-92, 4-1, 4-33, 4-49, 4-54, 4-56, 4-65, E-12, E-60, E-61

No-Action Scenario 5-1, 5-5, 5-6, 5-8

Nonproliferation S-13, 1-27, 1-30, 2-63, 2-65, 2-66, 2-67, 2-106, 2-107

Plutonium-239 S-1, S-7, S-9, S-10, S-14, 1-1, 1-12, 1-16, 1-21, 1-22, 1-24, 1-26, 1-31, 1-32, 2-9, 2-18, 2-20, 2-25, 2-27, 2-31, 2-61, 2-63, 2-64, 2-66, 2-68, 2-74, 2-76, 2-77, 2-79, 2-94, 2-96, 2-98, 2-103, 2-105, 4-31, 4-47, 4-50, 4-55, 4-58, 4-71, 5-2, A-8, A-9, A-12, B-4, C-10, C-23, E-5, E-6, E-8

Plutonium-242 S-6, S-7, S-9, 1-16, 1-20, 1-22, 1-24, 2-5, 2-6, 2-8, 2-9, 2-10, 2-59, 2-66, 2-68, 2-73, 2-76, 2-80, 2-92, 4-30, 4-34, 4-39, 4-47, 4-49, 4-54, 4-57, 4-70, 5-2, 5-4, A-8, A-9, B-4

Plutonium Working Group Report 1-13, 1-19

Processing to Metal S-3, S-9, 2-9, 2-24, 2-28, 2-32, 2-37, 2-39, 2-42, 2-61, 2-64, 2-66, 2-68, 2-80, 2-81, 2-92, 2-95, 2-96, 2-97, 2-98, 2-102, 2-103, 4-17, 4-30, 4-31, 4-32, 4-33, 4-39, 4-40, 4-47, 4-48, 4-49, 4-50, 4-54, 4-55, 4-57, 4-59, 4-60, 4-62, 4-63, 5-2

Processing to Oxide S-9, 2-6, 2-8, 2-9, 2-13, 2-15, 2-18, 2-26, 2-29, 2-32, 2-37, 2-38, 2-43, 2-59, 2-61, 2-64, 2-66, 2-68, 2-80, 2-81, 2-92, 2-93, 2-94, 2-95, 2-96, 2-97, 2-100, 4-17, 4-30, 4-31, 4-32, 4-33, 4-39, 4-40, 4-41, 4-47, 4-48, 4-49, 4-50, 4-54, 4-55, 4-57, 4-58, 4-59, 4-60, 4-62, 4-63, 5-2

Programmatic material S-6, 1-22, 2-5, 2-6, 4-1, A-8

Public nonradiological health 3-25

Public health 3-22, 3-26, 4-17, 4-46, 4-49, 5-4, 5-11, 8-1, 8-2

PUREX process 2-9, 2-32, 2-33, 2-42, 2-43, 2-107, C-10, C-12, C-18, C-27

Radiation dose 3-22, 3-27, 4-41, 4-43, 4-52, A-2, B-2, C-59, E-59

Radiation exposure 1-14, 1-15, 1-26, 3-22, 3-24, 3-26, 4-2, 4-39, 5-1, 5-2, B-2, B-3, B-5, E-9, E-56

RADTRAN 4-39, 4-71

Reactor Disassembly Basins S-5, 1-14, 1-16, 1-18, 1-30, 2-4, 2-11, 2-64, 2-81, 2-83, 2-104, A-2, A-8, A-13, C-5, E-58

Receiving Basin for Offsite Fuels (RBOF) S-2, S-7, 1-11, 1-15, 1-17, 1-18, 1-23, 1-24, 1-30, 2-11, 2-15, 2-31, 2-41, 2-77, 2-91, 2-98, A-3, A-8, A-12, A-13, A-15, C-28, C-29, E-57, E-61

Red-cockaded woodpecker 4-81

Risk S-1, S-3, S-4, S-5, S-12, 1-8, 1-9, 2-47, 2-83, 2-98, 2-103, 2-104, 2-107, 3-26, 4-18, 4-39, 4-63, 4-64, 4-65, 4-68, 4-69, 4-70, 4-71, 5-6, 5-13, B-1, B-3, B-5, C-44, E-2, E-3, E-6, E-8, E-12, E-55, E-59, E-60, E-70

Saltstone facility 2-90, C-56

Savannah River Technology Center S-7, 1-24, 2-89, A-13, C-14, C-27, C-29, C-30, E-58, E-60, E-67, E-68

Seismicity 3-4, 3-7

Spent Fuel Working Group Report 1-13, 1-14, 2-4

SRS Waste Management EIS 1-13, 1-28, 1-29, 5-14, 5-16

Stable material S-3, S-6, 1-13, 1-25, 2-1, 2-4, 2-5, 2-68, 2-83, 2-92, 4-49, 4-54, 5-2, 5-4, A-3, E-58, E-59, E-60

Threatened and Endangered Species 3-1, 4-3, 4-81, 8-6

TNX Research Facility E-60

Tornadoes 2-81, 3-13, 4-64, C-38, E-1, E-72

Transuranic waste 1-13, 1-19, 4-34, 4-56, 4-57, 4-58, 4-59, 4-60, 4-61, 4-62, 4-76, 5-8, 5-9, 5-16, C-49, C-53, C-59, C-61, C-65

Tritium S-1, 1-1, 1-12, 3-10, 3-11, 3-14, 5-11, A-13, B-4, C-1, C-61, C-67

Uranium S-7, S-9, S-10, S-13, 1-1, 1-15, 1-16, 1-17, 1-18, 1-20, 1-22, 1-24, 1-25, 1-27, 1-28, 1-30, 2-20, 2-25, 2-26, 2-27, 2-28, 2-29, 2-31, 2-32, 2-33, 2-36, 2-37, 2-38, 2-39, 2-41, 2-42, 2-43, 2-45, 2-46, 2-60, 2-64, 2-65, 2-68, 2-71, 2-73, 2-79, 2-80, 2-95, 2-96, 2-98, 2-100, 2-105, 2-106, 3-11, 3-25, 4-1, 4-3, 4-17, 4-31, 4-32, 4-33, 4-40, 4-47, 4-48, 4-50, 4-52, 4-53, 4-55, 4-59, 4-60, 4-71, 5-2, 5-4, 5-15, A-1, A-3, A-6, A-7, A-8, A-10, A-14, B-4, C-1, C-5, C-10, C-12, C-13, C-14, C-16, C-22, C-23, C-27, C-28, C-30, C-31, C-32, C-33, C-36, C-61, E-6, E-55, E-59, E-60

Vitrification Alternative 2-9, 2-22, 2-37, 2-60, 2-61, 2-64, 2-83, 2-93, 2-96, 4-17, 4-32, 4-33, 4-40, 4-47, 4-48, 4-49, 4-50, 4-51, 4-54, 4-55, 4-57, 4-58, 4-59, 4-60, 4-62, 4-63

Warm canyon 1-5, C-9, C-10, C-13, C-14, C-16, C-17, C-18, C-24, C-26, C-28

Wastewater 1-28, C-26, C-67

Weapons-usable fissile materials 1-26, 1-27, 1-31, 2-66, 2-104, 2-106, 5-11

Worker nonradiological health and safety 3-27

Worker radiological health 3-26, 4-30, 4-31, 4-32, 4-33

XOQDOQ 4-46

Y-12 Plant 1-27, 1-28